AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 100.002

Question:

10 CFR 52.47(a)(iv) requires proposed resolutions of certain safety issues which are identified in "the version of NUREG-0933 current on the date six months prior to application..." Please state the version of NUREG-0933 you are referencing for the AP1000 design certification application.

Westinghouse Response:

Section 1.9 of the DCD will be revised to the guidance of NUREG-0933, Supplement 25 issued June 2001. This is the version of NUREG-0933 that was in effect within 6 months of our application.

Design Control Document (DCD) Revision:

DCD Section 1.9 is revised as follows:

1.9.6 References

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," 1981.
2. Report NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982.
3. NUREG-0933, "A Prioritization of Generic Safety Issues," June ~~2000~~1.

PRA Revision:

None

RAI Number: 210.027

Question:

Reference, Volume 6, Section 3.9.5.2.1, Level A and B Service Conditions, Pg. 3.9-79:

The last bullet specifies "Earthquake" as a service condition for both Level A and Level B service limits. How is this earthquake defined? Is it included in both Level A and Level B load combinations? Earthquake loading does not appear to be included in any of the Level A or Level B loading combinations listed in Table 3.9-5.

Please provide clarification regarding earthquake loadings, other than SSE, and loading combinations considered in the design of the RPV core support structures.

Westinghouse Response:

AP1000 core support structures are designed for one occurrence of the safe shutdown earthquake which is evaluated as a Service Level D condition for pressure boundary integrity (See DCD section 3.7).

In DCD section 3.9.5.2.1, the "earthquake" listed is an earthquake that is included only in the fatigue evaluation of the structure. Typically, there are five seismic events considered with an amplitude equal to one-third of the safe shutdown earthquake response. Each of the one-third safe shutdown earthquake events has 63 high-stress cycles. See response to RAI 210.047.

Earthquake loading is not included under Service Level A or B in Table 3.9-5 because this "earthquake" is only considered for fatigue evaluation by the inclusion of the additional cycles as defined above to the thermal transient cycles given in DCD Table 3.9-1.

DCD section 3.9.5.2.1 will be modified to clarify this approach.

Design Control Document (DCD) Revision:

From DCD page 3.9-78:

3.9.5.2.1 Level A and B Service Conditions

The level A and B service conditions that provide the basis for the design of the reactor internals are:Fuel assembly and reactor internals weight

- Fuel assembly and core component spring forces, including spring preloading forces

- Differential pressure and coolant flow forces
- Temperature gradients
- Operational thermal transients listed in Table 3.9-1
- Differences in thermal expansion, due to temperature differences and differential expansion of materials
- Loss of load/pump overspeed
- Earthquake **(Included only in fatigue evaluation; amplitude equal to one-third of the safe shutdown earthquake response)**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.059

Question:

Reference, Volume 6, Section 3.9.4.2.1, Pg. 3.9-72, second bullet:

Please indicate how the 166.755 inch travel dimension is determined, and discuss the implication of the three significant figures used for the fractional part of the dimension specifying maximum travel.

Westinghouse Response:

The 166.755 inch travel dimension is the maximum control rod travel considering nominal, no tolerance stack-up considerations. This dimension is calculated based on the need for the control assembly absorber to travel through the length of the active fuel and for the tips of the absorber rods to remain engaged in the guide thimbles so that alignment between rods and thimbles is maintained when the assembly is withdrawn through its full travel.

The CRDM latch assembly includes movable gripper latches that engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-inch steps. The drive rod, which is coupled to the rod cluster control assembly, has a 5/8-inch pitch from groove to groove that engages the latches during holding or moving of the drive rod. Therefore, the control rod drive mechanisms move the control assembly in 5/8-inch or 0.625 (± 0.015)-inch steps. Thus, in decimal format, the CRDM travel distance is given to three significant figures.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.060

Question:

Reference, Volume 6, Section 3.9.4.2.1, Pg. 3.9-72, third bullet:

The 400-pound maximum load capability does not agree with the CRDM load capacity specified on Pg. 3.9-68, second paragraph. Please clarify.

Also, please identify the maximum total weight of the rod cluster control and drive rod assembly which is actually raised or lowered by each CRDM.

Westinghouse Response:

The AP1000 CRDM maximum load capacity is 400 pounds.

DCD section 3.9.4.1.1 will be modified to reflect the 400-pound maximum load capability.

The weight of the rod cluster control and drive rod assembly is approximately 320 pounds.

Design Control Document (DCD) Revision:

From DCD section 3.9.4.1.1:

The control rod drive mechanism withdraws and inserts a rod cluster control assembly or gray rod control assembly as shaped electrical pulses are received by the operating coils. An on or off sequence, repeated by silicon-controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Withdrawal of the drive rod and rod cluster control assembly or gray rod control assembly is accomplished by magnetic forces. Insertion is by gravity. The mechanism is capable of raising or lowering a **maximum 400**~~360~~-pound load (which includes the drive rod weight) at a rate of 45 inches per minute.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.065

Question:

Hydrodynamic loads could be significant for piping systems with fast opening and closing valves. The reactor vessel head vent system (RVHVS) piping has been identified as a system susceptible to such large hydraulic transient loadings. Experience from operating nuclear power plants indicates that thermal hydraulic loads could be very severe for certain operating modes if the piping layout does not take this aspect into careful consideration. Please provide assurance that the AP1000 plant specific RVHVS piping configuration can be designed to withstand the transient thrust forces caused by the valve opening without performing the thermal hydraulic and structural dynamic analyses at this time. To demonstrate the conceptual design and facilitate the staff's review, it is suggested that a sketch be provided along with the response.

NOTE: This question was discussed with Westinghouse representatives in a meeting at Westinghouse Energy Center held from Monday, September 9, 2002, through Wednesday, September 11, 2002.

Westinghouse Response:

The reactor vessel head vent piping systems (RVHVS) utilized in conventional Westinghouse PWRs consist of two basic layout configuration types. One layout configuration utilizes a modified CRDM housing that is capped at the top and a one inch pipe welded to it. The flow path from which the RPV venting would occur is through the modified CRDM (approximately equivalent to a three inch pipe), into the one inch head vent piping containing the one inch valves, and discharging into either the containment or the discharge header of the Pressurizer Safety and Relief Valve System. This layout configuration can result in significant thermal hydraulic loads (upwards of 30 to 40 kips) due to RPV through the modified CRDM and into the one inch piping.

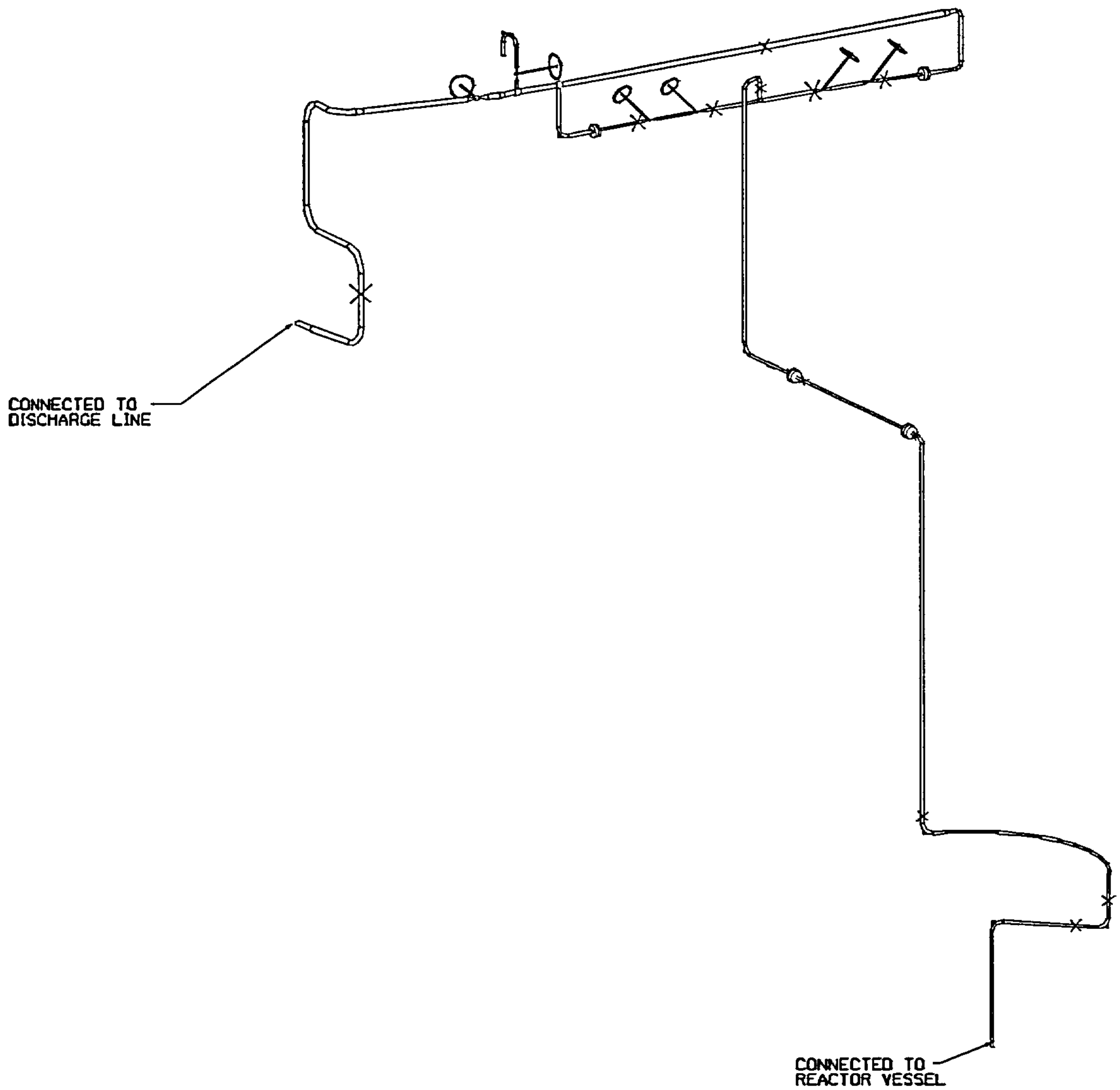
The second layout configuration type utilizes a one inch vent pipe welded to a dedicated one-inch penetration nozzle in the head of the RPV. For this layout configuration, the flow path for the venting is directly from the RPV into the one inch piping, through the one inch valves, and into either the containment or the PSARV discharge piping. This layout configuration significantly reduces the magnitude of the thermal hydraulic loads resulting from the venting of the RPV.

For both the AP600 and AP1000 RVHVS, the piping layout was developed such that the thermal hydraulic loads are kept to a minimum. The second layout configuration type as described above is utilized with the one inch piping welded directly to the RPV head. In addition, length of piping between elbows is also kept to a minimum in order to reduce the effect of these thermal hydraulic loadings. Long straight sections of piping (i.e. greater than 20 feet) are not used.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Piping layout drawings and isometrics for both the AP600 and AP100 Reactor Vessel Head Vent Systems were made available to the NRC staff for review at the meetings held on September 9th through the 11th at the Westinghouse office. The review confirmed that the layouts were developed to minimize the effect of severe thermal hydraulic loadings within the piping system. A simplified sketch of the AP1000 RVHV piping system is shown below.



REACTOR HEAD VENT PIPING

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.067

Question:

NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," and Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," requested nuclear power plant licensees to implement a program to ensure that erosion/corrosion does not lead to degradation of single-phase and two-phase high-energy carbon steel systems. The main feed water system was highlighted as an example of a pipe line that is susceptible to this type of degradation mechanism; also referred to as flow-accelerated corrosion (FAC). This type of problem should be prevented, to the extent practical, at the design stage to minimize the potential impacts. Please provide additional information discussing design measures for the AP1000 plant to minimize the effects of FAC on the feedwater line inside and outside the containment, including selection of material that is less susceptible to FAC, and component type and layout considerations.

Westinghouse Response:

The sizing and material selection of the feedwater line considers erosion/corrosion concerns associated with fluid velocity and chemistry. The safety related portion of the main feedwater piping, both inside containment and in the auxiliary building, is low alloy steel (SA-335, Gr.P-11). This material was selected to minimize the potential for erosion/corrosion.

The feedwater layout is designed with considerations to mitigate erosion damage. The design and layout of piping systems consider the effects on the piping material of fluid velocity, bend location, and potential flash points. Minimization of pipe fittings (tees, etc.) and flow stream disturbances, and the use of bends in lieu of elbows also mitigates erosion.

The AP600 main feedwater piping system was assessed using EPRI program "CHECMATE" which evaluates the erosion/corrosion resistance of piping system. The assessment considered water chemistry inputs for Ph levels of 9.0 and oxygen content of 2 ppb, and concluded the following for the SA-335 GR P-11 material:

a) No significant piping erosion occurred per year
b) Life expectancy to reach code minimum wall thickness was greater than 1000 years.

The layouts of the main feedwater piping systems are similar for the AP600 vs. the AP1000 designs, with the main difference being the increase in pipe size from 16 inch for the AP600 to 20 inch for the AP1000. The water chemistry characteristics and the fluid velocity (approximately 25 ft/sec) are also similar for both designs, therefore significant margin exists in life expectancy of the AP1000 main feedwater line.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.068

Question:

Reference: WCAP-15800, "Operational Assessment for AP1000," Revision 0, dated April 2002, Pgs. 2-2 and 2-3.

The resolution comment provided for Bulletin 81-01, "Failure of Mechanical Snubbers," and for Bulletin 81-01, Revision 1, "Surveillance of Mechanical Snubbers," refers to AP1000 DCD Section 3.9.6. However, DCD Section 3.9.6 addresses only inservice testing of pumps and valves, and does not include any information on mechanical snubbers. It is acknowledged that Bulletin 81-01 dealt with examinations of snubbers installed in operating plants, and this aspect of the bulletin is not applicable to the AP1000 design certification. But, the surveillance and qualification testing implications of these bulletins should be addressed during the design certification process. DCD Section 3.9.6 does not provide this information. Please provide additional discussion or references, including commitments to current documents which address surveillance and testing of mechanical snubbers used in the AP1000 design.

Westinghouse Response:

The test of dynamic restraints, i.e. snubbers, is governed by the ISI requirements of section XI of the ASME BPV Code. Subsection 3.9.3.4.3 of the DCD discusses requirements for the production and qualification of hydraulic snubbers. Subsection 3.9.8.3 identifies the requirement for the Combined License applicant to develop a program to verify the operability of snubbers utilized in the design of the AP1000. Additionally, subsection 5.2.4 defines that inservice inspection and testing of Class 1 components, including supports, are performed in accordance with Section XI of the ASME Code.

Design Control Document (DCD) Revision:

The last paragraph of subsection 3.9.3.4.3 will be revised as follows:

The operability of essential snubbers is verified by the Combined License applicant by verifying the proper installation of the snubbers, and performing visual inspections and measurements of the cold and hot positions of the snubbers as required during plant heatup to verify the snubbers are performing as intended. **The ASME OM Code used to develop the inservice testing plan for the AP1000 Design Certification is the 1995 Edition and 1996 Addenda. Inservice testing is performed in accordance with Section XI of the ASME Code and applicable addenda, as required by 10 CFR 50.55a.**

PRA Revision:

None

WCAP-15800 Revision:

Section 2 entitled "I.E. Bulletins" will be revised as follows:

I.E. BULLETIN		
Number	**Title**	**Comment**
88-01	Failure of Mechanical Snubbers (1/82)	Inservice Inspections in accordance with ASME Section XI DCD ~~3.9.6~~ **3.9.3.4.3**
88-01 R1	Surveillance of Mechanical Snubbers	Inservice Inspections in accordance with ASME Section XI DCD ~~3.9.6~~ **3.9.3.4.3**

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.069

Question:

Reference: WCAP-15800, Pg. 3-7.

The resolution comment provided for GL 80-109, "Guidelines for SEP and Soil Structure Interaction Reviews," states that the AP1000 has been designed for a range of soil conditions, and refers to DCD Section 3.7.2. Please provide a revised resolution comment to be consistent with the limitations placed on the AP1000 design for certification for hard rock sites only. (As stated in your letter dated February 13, 2002, ADAMS Accession No. ML020640065. This issue was also discussed in a telephone conference on August 22, 2002 between Westinghouse personnel and the NRC staff, in which it was concluded that the AP1000 DCD Section 3.7 would be revised to remove all discussions of seismic analysis using soil-structure interaction methodology.)

Westinghouse Response:

WCAP-15800 will be revised to correct the resolution statement.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

WCAP-15800 will be revised as shown:

80-109	Guidelines for SEP and Soil Structure Interaction Reviews (12/80)	AP1000 has been designed for a ~~range of soil conditions~~**hard rock site.** DCD Section 3.7.2

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.070

Question:

Reference AP1000 Design Control Document, Highlight/Strikeout Version, APP-GW-GL-701 Revision 0, dated January 2002, Section 1.9.4.2.3, "New Generic Issues."

The "AP1000 Response" to Issue 113, "Dynamic Qualification of Large-Bore Hydraulic Snubbers," provided on DCD page 1.9-61, refers to requirements established in "ASME OM Code - 1990," which is an older version of the OM code.

Please provide an exemption request for referencing an older issue of the ASME OM Code, or revise the statement to adopt the latest edition and addenda to the ASME OM Code incorporated by reference in 10 CFR 50.55a.

Westinghouse Response:

Westinghouse agrees with the above position. The AP1000 DCD will be revised as shown below.

Design Control Document (DCD) Revision:

AP1000 DCD response to Issue 113 will be revised as follows:

Issue 113 Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

AP1000 Response:

The AP1000 plant uses significantly fewer hydraulic snubbers than do currently operating plants. In addition to the recommendations in the draft regulatory guide, testing requirements have been established in ASME OM Code — ~~1990~~**1995 Edition up to and including the 1996 Addenda**, "Code for Operation and Maintenance of Nuclear Power Plants." Subsection 3.9.3.4.3 discusses requirements for production and qualification testing. The design of the hydraulic snubbers permits required preoperational and inservice testing.

Subsection 3.9.8.3 identifies the requirement for Combined License applicant information on snubber operability testing.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 220.001

Question:

In order to verify the design adequacy of AP600, Westinghouse conducted various performance tests for unique AP600 systems. The outcome of these tests was used to define the resulting loads for the design of seismic Category I structures (containment vessel, containment internal structures, and coupled shield building and auxiliary building). In the AP1000 DCD Section 1.5, "Requirements for Further Technical Information," Westinghouse states that the AP600 test results are also applicable to the AP1000, and cites Reference 25 [WCAP-15613, "AP1000 PIRT and Scaling Assessment," (Proprietary), WCAP-15706 (Nonproprietary), dated March 2001] for documentation of its evaluation to support this conclusion. DCD Table 1.5-1 provides a list of AP600 tests and AP1000 evaluations with references to test and evaluation documentation. However, the details of how design loads for structural evaluation are not evident from these reports. The design of AP1000 containment structure, the containment internal structures and other Category I structures (i.e., shield building, auxiliary building, containment air baffle, cable tray supports, and HVAC supports) needs to consider the effects of loads from thermal striping of the exterior surfaces of the taller AP1000 containment, the loads from the higher mass and energy release for AP1000 containment internal structures, and the loads that are applicable to the other Category I structures for AP1000. Therefore, please provide a detailed technical basis for the loads for the three types of structures discussed below.

A. loadings on the AP1000 Containment, due to thermal striping.

B. loadings on the AP1000 containment internal structures.

C. loadings on other AP1000 Category I structures (i.e., shield building, auxiliary building, containment air baffle, cable tray supports, and HVAC supports).

Westinghouse Response:

AP600 test results were used to evaluate design loads for the AP1000 structures as described and justified below:

- the AP600 Passive Containment Cooling System water distribution tests described in References 8, 9 and 10 listed in Table 1.5-1 were used to define wet and dry regions around the containment vessel to evaluate the potential for buckling of the vessel due to an asymmetric thermal distribution.

- the AP600 Automatic Depressurization System hydraulic tests described in References 3 and 4 listed in Table 1.5-1 were used to define design loads on the AP1000 in-containment refueling water storage tank constructed integrally with the containment internal structures.

- the AP600 wind tunnel tests described in References 11 through Reference 15 listed in Table 1.5-1 were used to define design loads on the AP1000 containment vessel and containment air baffle.

No other AP600 tests were used to define design loads on AP1000 structures.

A. Passive Containment Cooling System Water Distribution Test

The AP600 water distribution tests were used to define wet and dry regions around the containment vessel to evaluate the potential for buckling of the vessel due to an asymmetric thermal distribution. The AP1000 containment vessel head geometry is the same as that of the AP600 so the selection of wet and dry strips for the AP600 evaluation is directly applicable to AP1000.

The AP600 containment vessel evaluation for asymmetric temperature distribution was found acceptable as stated in the AP600 FSER. The evaluation is documented in Westinghouse's letter of July 02, 1993 (ET-NRC-93-3916) which was provided in response to NRC's request for additional information dated 4/2/93. The conclusion of the evaluation states:

> "The containment vessel was evaluated for temperature variations around the vessel that were conservatively postulated based on a review of the water distribution tests and other safety analyses. Shell stresses due to the thermal loads were conservatively evaluated and demonstrated large margin against buckling. The evaluation has demonstrated that such temperature variations are not significant to the design of the containment vessel."

The thickness of the AP1000 containment vessel head is 1.625 inches and is the same as the AP600. The thickness of the AP1000 containment vessel cylinder is 1.75 inches compared to the 1.625 inch thickness of the AP600. This increase in thickness will increase the margin against buckling. Thus, the conclusions of the AP600 evaluation are also applicable to the AP1000 and the temperature variations are not significant to the design of the containment vessel.

B. Automatic Depressurization System Hydraulic Tests

The AP600 Automatic Depressurization System hydraulic tests were used to define design loads on the AP1000 in-containment refueling water storage tank constructed integrally with the containment internal structures. Two pressure time histories, characterized by different shapes and frequency content, were selected from the tests as representative of the sparger discharge pressures.

- A pressure time history representing two phase blowdown with high mass flow rates (above 650 lbs/sec) and quality resulting in significant frequency content below 50 Hertz and pressure peaks at the test tank walls comparable to the highest values measured during the two phase blowdown tests.

- A pressure time history representing opening of the second or third stage of ADS at full pressure. This is characterized by pure steam flow.

The time histories from these tests are also applicable to the AP1000 since they occur at the beginning of the transient, and the automatic depressurization system and the initial conditions are the same for the two plant designs. The design of the automatic depressurization system valves that discharge into the in-containment refueling water storage tank are the same for both plants, including the key features controlling the blowdown, such as valve opening times, flow areas, flow rates and fluid conditions. The design of the sparger and the in-containment refueling water storage tank are also the same for both plants.

The response of the AP1000 in-containment refueling water storage tank to these time history forcing functions is addressed in Westinghouse's response to RAI 220.009.

C. Passive Containment Cooling System Wind Tunnel Tests

The AP600 wind tunnel tests were used to define design loads on the AP1000 containment vessel and containment air baffle.

The containment vessel is surrounded by the shield building and is only subjected to wind loading in the upper annulus. The containment air baffle is located within the annulus between the containment vessel and the shield building. It interfaces with the passive containment cooling system and separates downward flowing air entering at the air intake openings at the top of the cylinder portion of the shield building from upward flowing air that cools the containment vessel and flows out of the discharge diffuser.

The wind loading for seismic Category I structures is in accordance with American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-98. The analytical procedures given in ASCE 7-98, Section 6.5 do not apply to the differential pressures within the annulus and on the containment air baffle since this structure has unusual, irregular geometric shapes, and is subject to channeling effects. ASCE 7-98 permits wind tunnel tests in lieu of analytical procedures for such a configuration.

Westinghouse performed wind tunnel tests for the AP600 configuration to confirm functionality of the passive air-cooling under wind conditions. As noted in subsection 3.3.2.4 of the AP1000 Design Control Document, the pressure coefficients from these tests are used to define wind and tornado loadings on the containment vessel and containment cooling air baffle.

The AP600 wind tunnel tests developed pressure coefficients (C_P) for the containment air baffle area. The envelope C_P values reflect the winds from different directions (azimuths). To obtain the pressure loading, the dynamic pressure at roof height is multiplied by the pressure coefficient at the different levels of the containment air baffle.

The AP1000 shield building and containment vessel are ~25 feet higher than the AP600. The containment air baffle is similar to the AP600 plant except that it is also 25 feet longer. The AP1000 air intakes are 12 feet wide by 6 feet 6 inches high and have approximately the same flow area as the 16 feet wide by 5 feet high AP600 air intakes. The discharge for both plants is 32 feet in diameter. The air intakes and discharge are 25 feet higher but are situated around the shield building at the same radius as the AP600 plant. The pressure coefficients will not change since the basic shape (geometric design) and channeling effects are similar to the AP600 plant. Thus, it is concluded that the pressure coefficients (C_P) obtained from the wind tunnel test data for the AP600 plant are also applicable to the AP1000 plant.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 220.002

Question:

For the AP600 containment cylindrical shell, the nominal design thickness is 1.625". However, for the bottom cylinder section, Westinghouse increased the shell thickness to 1.75" in order to "provide margin in the event of corrosion in the embedment transition region" (quote from AP600 DCD). For the AP1000 containment cylindrical shell, the nominal design thickness is a uniform 1.75" for the entire length. The 1.75" thickness just meets the minimum thickness requirements (1.7455") of the 1998 American Society of Mechanical Engineers (ASME) Code Section III, Subsection NE, Paragraph NE-3324.3(a), based on 59 pounds-per-square inch (psi) design pressure, 300°F design temperature, S = 26.4 ksi (thousand pounds-per-square inch), and R = 780". There is no margin in the nominal design thickness for corrosion allowance. Therefore, Westinghouse is requested to provide a technical justification for:

A. eliminating the corrosion allowance for the embedment transition region (deviation from AP600 design philosophy), and

B. making no provision for general corrosion of the containment shell over its 60 year design life in defining the nominal design thickness. Paragraph NE-3121 specifically addresses corrosion allowance for Class MC components.

Westinghouse Response:

A. As stated in the question, Westinghouse increased the shell thickness to 1.75" for the bottom cylinder section in order to "provide margin in the event of corrosion in the embedment transition region". For the AP1000, all of the cylinder section has been increased to 1.75" to accommodate the increase in design pressure. The thickness in the bottom section was not increased further since a greater thickness would require post weld heat treatment.

The minimum required thickness of the cylinder in accordance with NE-3324.3(a) is 1.726" rather than the 1.7455" stated in the question. This is based on the effect of temperature on allowable stress intensity which is incorrect in the 1995 and 1998 editions of the ASME Code. It has been corrected in the 2002 Addenda. The allowable stress intensity for Class I is 24.3 for all temperatures up to 500 °F. Thus, the allowable stress intensity for Class MC is 26.7 ksi rather than 26.4 ksi. The minimum design thickness for AP600 is 1.597" and the thickness provided is 1.625". Thus the margin for AP1000 is 0.024" compared with 0.028" for AP600.

The containment vessel would be acceptable even if some corrosion were to occur in the transition region. Evaluation would be performed to confirm the structural integrity in accordance with paragraph ASME Section XI, Paragraph IWE-3122.3. The transition region is close to the concrete embedment outside the vessel at elevation 100'. Shell stresses are shown in DCD Figure 3.8.2-5 for the internal design pressure of 59 psig. Table 220.002-1 shows these stresses for locations within 20 feet of the base. The allowable membrane stress intensity away from the discontinuity is 26.7 ksi. Close to the discontinuity the stresses are evaluated as local primary membrane stresses with a 50% higher allowable stress intensity of 40.0 ksi. The maximum stress is 27.30 ksi and occurs at an elevation of 110.0'. This provides significant margin for evaluation of potential corrosion in the transition region.

B. Corrosion protection is provided by coating the vessel as described in DCD subsection 3.8.2-8 and 6.1. Additional margin is provided as described in the response to Part A of this question.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Table 220.002-1

Containment Vessel Shell Membrane Stresses

Design Internal Pressure of 59 psig

ELEM	Elev. (ft)	Meridional Stress (ksi)	Circumferential Stress (ksi)	Stress Intensity (ksi)
1	100.00	14.20	4.25	14.20
2	101.02	14.20	2.84	14.20
3	102.05	14.20	1.68	14.20
4	103.09	14.19	3.57	14.19
5	104.13	13.16	8.49	13.16
6	105.10	13.15	14.56	14.56
7	106.08	13.15	19.85	19.85
8	107.06	13.15	23.61	23.61
9	108.04	13.15	25.87	25.87
10	109.02	13.15	26.96	26.96
11	110.00	13.15	27.30	27.30
12	111.00	13.15	27.22	27.22
13	112.00	13.15	26.97	26.97
14	113.00	13.15	26.71	26.71
15	114.04	13.15	26.49	26.49
16	115.08	13.15	26.34	26.34
17	116.12	13.15	26.27	26.27
18	117.16	13.15	26.24	26.24
19	118.20	13.15	26.25	26.25
20	119.24	13.15	26.29	26.29
21	120.27	13.15	26.36	26.36

RAI Number: 251.013

Question:

Because temperature affects the neutron embrittlement of the materials, provide information on the cold leg temperature. If a plant will operate at a cold leg temperature below 274 degrees C (525 degrees F), discuss the effects of temperature on embrittlement of reactor vessel materials. This question was asked for the AP600 review (AP600 RAI 252.84). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD. (Section 5.3.2).

Note: AP600 RAI 252.84 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 8, 1993 (NUDOCS Accession No. 9301130165).

Westinghouse Response:

The AP1000 cold leg temperature exceeds 525 F under normal steady state conditions. The minimum steady state cold leg temperature is 535.0 F. This value corresponds to the conditions of 100% power, Thermal Design Flow and 10% tube plugging.

DCD section 5.3.3.1 will be modified to incorporate the minimum AP1000 steady state cold leg temperature.

Design Control Document (DCD) Revision:

From DCD page 5.3-13:

5.3.3.1 Limit Curves

Heatup and cooldown pressure-temperature limit curves are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section III of the ASME Code are employed in the analysis of protection against nonductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil ductility temperature, which includes a reference nil ductility temperature shift (ΔRT_{NDT}), initial RT_{NDT} and margin. The extent of the RT_{NDT} shift is enhanced by certain chemical elements (such as copper and nickel).

Predicted ΔRT_{NDT} values are derived considering the effect of fluence and copper and nickel content for the reactor vessel steels exposed to 550°F temperature. U.S. NRC Regulatory Guide 1.99 is used in calculating adjusted reference temperature. **Since the AP1000 cold leg temperature exceeds 525 F (minimum steady state temperature is 535 F at 100% power, thermal design flow, and 10% tube plugging) the procedures of Regulatory Guide 1.99 for nominal embrittlement apply.** The heatup and cooldown curves are developed considering a sufficient magnitude of radiation embrittlement so that no unirradiated ferritic materials in other components of the reactor coolant system will be limiting in the analysis.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.014

Question:

Describe the lead factors for surveillance capsules. This question was asked for the AP600 review (AP600 RAI 252.96). The staff is requesting that a similar commitment be made in the AP1000 DCD that an analysis will be performed for the combined operator license application with the capsule/holder modeled in order to more accurately define the surveillance capsule lead factors and azimuthal locations. (Section 5.3.2)

Note: AP600 RAI 252.96 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

Westinghouse Response:

The surveillance capsule lead factor (LF) is defined as:

LF = [The Surveillance Capsule Fast Neutron Flux (E>1.0 MeV)] /
[The Pressure Vessel Maximum Fast Neutron Flux (E>1.0 MeV)]

The azimuthal locations of the capsule/holder have been defined in order to have a surveillance capsule leading factor ranging from 1.2 to 3.0 in front of the active core length. This corresponds to an azimuthal location between 12.25° and 15.83° with reference to 0° fixed at the center of the reactor core flats.

The lead factors are based on AP1000 DOORS 3.2 R-Theta and R-Z reactor vessel fluence and internals heat generation rate analyses. Reactor pressure vessel fast flux distribution has been evaluated by a 3D synthesis to properly evaluate the impact on the reactor vessel of the core shroud geometry.

The following assumptions have been utilized to define the surveillance capsule/holder location:

- The pressure vessel maximum fast neutron flux = 4.946E+10 n/cm^2-sec.
- The radial location of the pressure vessel maximum fast neutron flux is 199.7 centimeters (pressure vessel inner surface).
- The azimuthal locations of the pressure vessel maximum fast flux are 0° about the center of the reactor core flats.
- The surveillance capsule center is at a radius of 73.31 inches.
- A reference flat axial profile has been used in the R-Z calculations.

- Both the R-Theta and R-Z models do not include the surveillance capsule or the capsule holder geometry, but since standard plant analyses show a 25 percent increase in fast neutron flux with the surveillance capsule/holder modeled, a correction factor of 1.25 has been considered in the definition of the azimuthal position of the surveillance capsules.

While the above approach provides a good degree of accuracy for the preliminary definition of the capsule/holder location, an analysis will be performed for the Combined Operating License application with the capsule/holder modeled in order to confirm the proposed surveillance capsule lead factors and azimuthal locations.

The commitment to perform an analysis for the Combined Operating License application with the capsule/holder modeled will be included in DCD section 5.3.6.

Design Control Document (DCD) Revision:

From DCD page 5.3-22:

5.3.6 Combined License Information

5.3.6.1 Pressure-Temperature Limit Curves

The pressure-temp. curves shown in Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. However, for a specific AP1000, these curves will be plotted based on material composition of copper and nickel. Use of plant-specific curves will be addressed by the Combined License applicant during procurement of the reactor vessel. As noted in the bases to Technical Specification 3.4.15, use of plant specific curves requires evaluation of the LTOP system. This includes evaluating the setpoint pressure for the RNS relief valve.

5.3.6.2 Reactor Vessel Materials Surveillance Program

The Combined License applicant will address a reactor vessel reactor material surveillance program based on subsection 5.3.2.6.

5.3.6.3 Surveillance Capsule Lead Factor and Azimuthal Location Confirmation

The Combined License Applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

5.3.6.34 Reactor Vessel Materials Properties Verification

The Combined License applicant will address verification of plant-specific belt line material properties consistent with the requirements in subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3.

5.3.6.45 Reactor Vessel Insulation

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention.

From DCD page 1.8-13, Table 1.8-2:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
4.3-1	Changes to Reference Reactor Design	4.3.4
4.3-2	Fixed Incore Detectors	4.3.4
4.4-1	Changes to Reference Reactor Design	4.4.7
5.2-1	ASME Code and Addenda	5.2.6.1
5.2-2	Plant Specific Inspection Program	5.2.6.2
5.3-1	Reactor Vessel Pressure - Temperature Limit Curves	5.3.6.1
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2
5.3-3	**Surveillance Capsule Lead Factor and Azimuthal Location Confirmation**	**5.3.6.3**
5.3-~~3~~4	Reactor Vessel Materials Properties Verification	5.3.6.~~3~~4
5.3-~~4~~5	Reactor Vessel Insulation	5.3.6.~~4~~5
5.4-1	Steam Generator Tube Integrity	5.4.15
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1
6.1-2	Coating Program	6.1.3.2
6.2-1	Containment Leak Rate Testing	6.2.6
6.3-1	Containment Cleanliness Program	6.3.8.1
6.4-1	Local Toxic Gas Services and Monitoring	6.4.7
6.4-2	Procedures for Training for Control Room Habitability	6.4.7
6.6-1	Inspection Programs	6.6.9.1
6.6-2	Construction Activities	6.6.9.2
7.1-1	Setpoint Calculations for Protective Functions	7.1.6
8.2-1	Offsite Electrical Power	8.2.5

8.2-2	Technical Interfaces	8.2.5
8.3-1	Onsite Electrical Power	8.3.3
8.3-2	Onsite Electrical Power Plant Procedures	8.3.3
9.1-1	New Fuel Rack	9.1.6
9.1-2	Criticality Analysis for New Fuel Rack	9.1.6
9.1-3	Spent Fuel Racks	9.1.6
9.1-4	Criticality Analysis for Spent Fuel Racks	9.1.6
9.1-5	Inservice Inspection Program of Cranes	9.1.6

PRA Revision:

None

RAI Number: 251.016

Question:

Section 5.3.3.1 of the DCD indicates that the results of the material surveillance program will be used for the development of heatup and cooldown curves. Verify that the material surveillance program data that will be used for recalculating these curves is the plant specific data obtained by each combined operating licensee. (Section 5.3.3)

Westinghouse Response:

The heatup and cooldown curves will be recalculated based on plant specific material surveillance data obtained by each combined operating licensee.

As described in DCD section 5.3.6 the combined license applicant will first develop plant-specific heatup and cooldown curves during procurement of the reactor vessel, and the applicant will also address a reactor vessel material surveillance program based on DCD section 5.3.2.6. In addition to the plant specific surveillance data, surveillance program results from any data that demonstrates the embrittlement trends for the limiting beltline material will be considered in updates to the heatup and cooldown curves. These data could come from test reactors or from surveillance programs at other plants.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.017

Question:

Provide the details for the pressure-temperature limit calculations, including assumptions and margins. Identify any deviations from the recommended calculational procedures in Section 5.3.2 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis reports for Nuclear Power Plants." This question was asked for the AP600 review (AP600 RAI 252.105). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD (Section 5.3.3)

Note: AP600 RAI 252.105 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

Westinghouse Response:

The methodology given in Regulatory Guide 1.99, Revision 2, is used to estimate the shift in reference temperature.

The heatup and cooldown limit curves were calculated using the most limiting value of the adjusted reference temperature (ART). The methodology used to calculate the ART is given in the Regulatory Guide 1.99, Revision 2. A description of the methodology and assumptions used in the analysis follows:

$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$

where

Initial RT_{NDT} = reference temperature of the unirradiated material

$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10 \log f)}$

CF = chemistry factor, a function of copper and nickel content

$f = f_{surf} * (e^{-0.24X})$

f_{surf} = neutron fluence at the inner wetted surface of the vessel (10^{19} n/cm^2, E > 1.0 MeV)

X = depth into the vessel wall measured from the vessel inner surface (inches)

$\text{Margin} = 2 * [(\sigma_I)^2 + (\sigma_\Delta)^2]^{1/2}$

σ_I = standard deviation for the initial RT_{NDT}

σ_Δ = standard deviation for ΔRT_{NDT}

The assumptions used in calculating the 1/4T and 3/4T ARTs at the end of the 60-year design life (54 EFPY) for the forging and lower girth weld are as follows:

1) f_{surf} for the forging is 9.762×10^{19} n/cm^2 (E > 1.0 Mev).
2) f_{surf} for the lower girth weld is 2.847×10^{19} n/cm^2 (E > 1.0 Mev).
3) reactor vessel thickness at the beltline region is 8 inches.
4) weight percent Cu for the forging and lower girth weld metal is 0.03%.
5) weight percent Ni for the forging and lower girth weld metal is 0.85%.
6) The initial RT_{NDT} for the forging is -10°F and for the lower girth weld metal is -20°F.

The shift in the reference temperature at 1/4T and 3/4T for the forging weld metal is discussed below.

The "Margin" is the quantity, °F, that is to be added to obtain conservative, upper-bound values of adjusted reference temperature. If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If a generic value is used, σ_I should be obtained from the same set of data. The standard deviations from Regulatory Guide 1.99, Revision 2, for ΔRT_{NDT}, σ_Δ is 17° F for base metal and forgings and 28° F for weld metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}. The value of the initial RT_{NDT} is an estimated value. For this reason, a value of 17° F is used for σ_I, for both the forging and the weld metal.

Thus, the following margins were used in the calculations:

M = 45°F @ 1/4T location of the forging
M = 42°F @ 3/4T location of the forging
M = 66°F @ 1/4T location of the lower girth weld
M = 50°F @ 3/4T location of the lower girth weld

The calculated end-of-life (54 EFPY) 1/4T and 3/4T ARTs for the forging and weld metal are as follows:

ART @ 1/4T of forging = 63°F
ART @ 3I4T of forging = 56°F
ART @ 1/4T of lower girth weld = 93°F
ART @ 3/4T of lower girth weld = 66°F

The AP1000 heatup and cooldown curves were generated using the most limiting Adjusted Reference Temperature values and the NRC approved methodology documented in Reference 1, which is in accordance with the USNRC Standard Review Plan, with exception of the following:

1) The fluence values used are calculated fluence values (i.e. comply with Regulatory Guide 1.190), not the best estimate fluence values.
2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588).
3) The 1996 Version of Appendix G to Section XI is used rather than the 1989 version.
4) The flange requirement is also not considered per Reference 2.

Mechanics of the OPERLIM code, which is used to develop the allowable pressure-temperature relationships for normal heatup and cooldown rates, including criticality limits, are contained in Reference 3.

References:

1. Westinghouse Proprietary Class 3 Report "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A.
2. Westinghouse Proprietary Class 3 Report "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," WCAP-15315.
3. Westinghouse Proprietary Class 2 Report "Documentation and Verification of the OPERLIM Computer Code," WCAP-9186.

Design Control Document (DCD) Revision:

See response to RAI 251.018.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.018

Question:

Demonstrate that the pressure-temperature limits are in accordance with Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. For example, verify that the limit for the closure flange is satisfied. This question was asked for the AP600 review (AP600 RAI 252.106). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD. (Section 5.3.3)

Note: AP600 RAI 252.106 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

Westinghouse Response:

The AP1000 heatup and cooldown curves were generated using the most limiting Adjusted Reference Temperature values and the NRC approved methodology documented in Reference 1 with exception of the following:

1) The fluence values used are calculated fluence values (i.e. comply with Regulatory Guide 1.190), not the best estimate fluence values.
2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588).
3) The 1996 Version of Appendix G to Section XI is used rather than the 1989 version.

As stated in DCD section 5.3.3.1 the flange requirement is also not considered per Reference 2. The flange requirements were developed using the K_{Ia} fracture toughness. Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves. Reference 2 demonstrates that use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated.

References:

1. Westinghouse Proprietary Class 3 Report "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A.
2. Westinghouse Proprietary Class 3 Report "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," WCAP-15315.

Design Control Document (DCD) Revision:

From DCD page 5.3-13, Section 5.3.3.1:

The pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (EFPY) consistent with an expected plant design life of 60 years with 90 percent availability. Copper, nickel contents and initial RT_{NDT} for materials in the reactor vessel beltline region and the reactor vessel flange and the closure head flange region are shown in Tables 5.3-1 and 5.3-3. The operating curves are developed **with the methodology given in Reference 6 which is in** accordance with 10 CFR 50, Appendix G with the **following exceptions:**

1) The fluence values used are calculated fluence values (i.e. comply with Regulatory Guide 1.190), not the best estimate fluence values.

2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588).

3) The 1996 Version of Appendix G to Section XI is used rather than the 1989 version.

4) ~~that~~ Tthe flange requirement is not considered per Reference ~~6~~7.

The curves are applicable up to 54 effective full-power years. These curves, shown in Figures 5.3-2 and 5.3-3, are generic curves for the AP1000 reactor vessel design and they are limiting curves based on copper and nickel material composition. These curves are applicable as long as the following criteria are met:

- 10 CFR 50, Appendix G as related to pressure-temperature remains unchanged,
- Adjusted Reference Temperatures at 1/4T and 3/4T locations remain below the bases of Figures 5.3-2 and 5.3-3

From DCD page 5.3-22:

5.3.7 References

1. ASTM E-185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.”

2. Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” United States Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March, 2001.

3. WCAP-15557, “Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology,” S. L. Anderson, August 2000.

4. NRC Policy Issue, “Pressurized Thermal Shock,” SECY-82-465, November 23, 1982.

5. Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

6. **WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et. al., January 1996.**

7. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," W. Bamford, et. al., October 1999.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.019

Question:

Provide the fluence value that was used in calculating the RT_{PTS} for end-of-life. (Section 5.3.4)

Westinghouse Response:

The fluence values used in calculating the RT_{PTS} for end-of-life are:

Beltline Forging	9.762E19 n/cm^2
Beltline Girth Weld	2.847E19 n/cm^2

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.020

Question:

Provide a basis for not providing inspections, test, analyses, and acceptance criteria (ITAAC) related to the reactor coolant pump (RCP) flywheel fatigue analysis in Table 2.1.2-4. (Section 5.4.1)

Westinghouse Response:

The key requirements for the reactor coolant pump flywheel are to help provide the rotating inertia to ensure an adequate reactor coolant pump coastdown and for the flywheel to maintain its integrity at 125% of operating speed. The AP1000 includes two related design commitments for the reactor coolant pump flywheel (See DCD Tier 1 Table 2.1.2-4): 1) The RCPs have a rotating inertia to provide RCS flow coastdown on loss of power to the pumps; and 2) Each flywheel assembly can withstand a design overspeed condition.

In the design of the reactor coolant pump there are many calculations and analyses performed to verify the adequacy of the design of pump internal components. These calculations and analyses do not involve key safety measures and are thus are not included in the ITAACs. Fatigue of the flywheel is not a significant concern because the flywheel operates in the bearing water environment that is isolated from the primary coolant by the thermal barrier, resulting in insignificant fatigue usage for the flywheel.

In previous NRC reviews of other plant designs, reactor coolant system ITAACs without design commitments related to the flywheel fatigue analysis were found acceptable. Both the AP600 and the System 80+ received design certification with reactor coolant pump flywheel ITAACs related only to rotating inertia and flywheel integrity at 125% of operating speed, similar to the AP1000 ITAACs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.006

Question:

Section 5.4.2.4.2 indicates that tubes can be supported by either an open lattice design called eggcrates, or by a support plate design. The seventh paragraph of section 5.4.2.3.3 discusses tube supports only in terms of broached hole support plate design. Please clarify. (Section 5.4.2)

Westinghouse Response:

As indicated in DCD section 5.4.2.2 the support plate design is the AP1000 base design which is described in the DCD. The open lattice (eggcrate) design is mentioned because it is a possible option for the tube support design. The steam generator design descriptions and evaluations in DCD section 5.4.2 are based only on the broached hole support plate design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 252.007

Question:

Section 5.4.2.4.1 of the AP600 SSAR was revised in response to a staff question on archival material to indicate that a minimum of seven feet of tubing in the final heat treat condition is supplied. This information was deleted from the AP1000 DCD. Please address the standard or criteria that will be used to specify minimal tube archive requirements. (Section 5.4.2)

Westinghouse Response:

A minimum of seven feet of steam generator tubing in the final heat treat condition will be supplied as archival material.

This requirement will be included in DCD section 5.4.2.4.1.

Design Control Document (DCD) Revision:

From DCD 5.4-17, Section 5.4.2.4.1:

The heat and lot of tubing material for each steam generator tube is recorded and documented as part of the quality assurance records. Archive samples of each heat and lot of steam generator tubing material are provided to the Combined License applicant for use in future materials testing programs or as inservice inspection calibration standards. **A minimum of seven feet of tubing in the final heat treat condition is supplied.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.012

Question:

DCD, Tier 2, Section 14.2.9.1.1, "Containment Hydrogen Control Testing," contains general test acceptance criteria and methods for pre-operational testing of the passive autocatalytic recombiners (PARs) in the AP1000 design and Table 14.3-1, "ITAAC Screening Summary," lists an Inspection, Test, Analysis and Acceptance Criteria (ITAAC) for the Containment Hydrogen Control System. However, based on Revision 0 of the AP1000 DCD, reference and design feature information regarding PARS that was contained in the AP600 DCD has been deleted and is no longer present in AP1000 DCD Section 14.3, "Certified Design Material." Specifically, Westinghouse has deleted (1) the reference to Table 6.2.4.2 from Table 14.3-6, "Probabilistic Risk Assessment," and (2) the reference to Section 6.2.4.2.2 from Table 14.3-8, "Severe Accident Analysis." Westinghouse also removed the 3rd PAR in the IRWST from the AP1000 design. In the AP600 design, the third PAR is used to mitigate hydrogen combustion events that could possibly damage or prevent the IRWST from performing its intended safety function. (See RAI 480.001 for additional technical details.) Please provide additional information justifying deletion of the PARs from the AP1000 DCD.

Westinghouse Response:

Westinghouse prepared the AP1000 DCD in anticipation of the revised 10 CFR 50.44. The proposed 10 CFR 50.44 that has been published in the Federal Register (FR Volume 67, No. 149, pg. 50374) for comments permits the systems, components and instrumentation that perform the post-accident hydrogen monitoring functions to be classified as non-safety related. In addition, the hydrogen control systems whose function is based on controlling hydrogen buildup for design basis accidents may be eliminated from the plant design. Only those control systems required to protect the containment from severe accidents must be retained.

As a result, Westinghouse has changed the safety classification of the AP1000 PARs from AP1000 Equipment Class C (safety-related) to AP1000 Equipment Class D (nonsafety-related). Consequently, the DCD section 6.2 was revised consistent with the proposed requirements of 10 CFR 50.44, and a design basis analysis demonstrating PAR performance was not included in the DCD. Such analysis provided the basis for the inclusion of a PAR depletion rate in the AP600 DCD section 6.2.4.2. This depletion rate was deleted from the AP1000 DCD, and therefore is not included in Table 14.3-6.

Please see the response to RAI 480.001 regarding the inclusion of a PAR in the IRWST vent.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 280.001

Question:

Section 9.5.1.2.1.1 and Item 55 of Table 9.5.1-1 of the AP1000 Design Control Document (DCD) notes that the stairwells outside of primary containment serving as escape routes, access for firefighting, or access routes to areas containing equipment necessary for safe shutdown have not been enclosed in masonry or concrete towers with a minimum fire rating of 2 hours as specified in Position C.5.a.6. of CMEB 9.5.1. This includes fire areas: 1201 AF 01, 1202 AF 01, 1202 AF 05, 1204 AF 02, 1205 AF 01, 2000 AF 02, 2009 AF 01, 2003 AF 02, 4001 AF 01, 4001 AF 01, 4002 AF 02, and 4003 AF 02. The staff previously granted Deviation 9.5.1-2 for the use of gypsum stair towers in lieu of concrete or masonry for the AP600 in NUREG-1512, "Final Safety Analysis Report [FSER] Related to Certification of the AP600 Standard Design," on the basis that there were no missile hazards in the vicinity of the subject stairwells. External missile hazards were not considered in the staff's original evaluation of the AP600 stairwells. Following the events of September 11, 2001, the Federal Emergency Management Agency (FEMA) issued report FEMA 403, World Trade Center Building Performance Study: Data Collection, Preliminary Observations and Recommendations, dated May 2002. Based on the performance of the gypsum stairwell enclosures in the World Trade Center following the aircraft impacts, Section 8.2.2.1 of the FEMA report recommends the use of impact-resistant enclosures around egress paths, such as stairwells. In light of the potential for external missile hazards, such as aircraft, the staff has re-considered it's previous acceptance of gypsum stairwell enclosures in lieu of the concrete or masonry enclosure specified in the Branch Technical Position (BTP). Gypsum enclosures, while providing adequate fire resistance capability, are generally not considered impact-resistant to missiles. Considering the concern about the performance of gypsum stairwell enclosures to potential missile hazards please perform a detailed evaluation of the vulnerability of each of the AP1000 stairwells located outside containment that serve as escape routes, access for firefighting, or access routes to areas containing equipment necessary for safe shutdown to external missile hazards. For those stairwells that are potentially vulnerable to external missiles provide a revision to the DCD to incorporate the original BTP guidance for the use of concrete or masonry enclosures. For those stairwells that are determined not to be vulnerable to external missiles provide a technical basis for that determination.

Westinghouse Response:

DCD Section 3.5 addresses Missile Protection for the AP1000. The AP1000 is designed for protection of externally generated missiles in accordance with the criteria provided in the Standard Review Plan, Section 3.5.1.4. Westinghouse agrees that stairwells outside of primary containment serving as escape routes, access for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be adequately protected from external missiles. Westinghouse has performed a review of the protection provided for each applicable stairwell. The following is an assessment of the vulnerability of stairwells to external missiles.

Fire area 1201 AF 01 is also called stairwell S02. The top entry to this stairwell is located above grade on Level 3 (El. 100'-0"). The roof and all four sides of this stairwell entry are 2'-0" thick reinforced concrete. The Turbine Building is located adjacent to the 2'0" thick reinforced concrete Auxiliary Building Wall 11. The Turbine Building is a structural steel frame structure which is L306'-0" x W120'-0" x H124'-0". The balance of this stairwell is located below grade and consists of concrete structural walls on two sides and non-masonry wall on the other two sides. The external wall (Wall 11) of this stairwell that is below grade is a 3'-0" reinforced concrete wall and Wall P which serves as the east wall of the stairwell is 2'-0" thick. The inside walls that separate the stairwell from the two adjacent corridors are non-masonry walls. Additional protection for this stairwell is not required. The existing Auxiliary Building structure and the adjacent structure provides adequate protection against external missiles. An alternate access/egress route (1202 AF 01, stairwell S01) is also available in the event that this stairwell becomes inaccessible.

Fire area 1202 AF 01 is also called stairwell S01. It has modular steel stairs enclosed by concrete structural walls on two sides and non-masonry wall on the other two sides. The concrete walls are 2'-0" thick above grade and 3'-0" thick below grade. The non-masonry walls are the inside walls that separate the stairwell from the corridors and the adjacent elevator shaft. The interior spaces are protected by the above grade reinforced concrete shell of the auxiliary building. Stairwell S02 (1201 AF 01) provides an alternate to Stairwell S01 as the route from Level 3 (El. 100'-0") to the two floor levels below grade (Level 1 and Level 2). Stairwell S05 (1202 AF 05) provides an alternate to Stairwell S01 as the route from Level 3 (grade level) to the Main Control Room (Level 4). Protection is also provided by the Annex Building, which is located adjacent to the 2'-0" thick reinforced concrete Auxiliary Building structural Wall I. The Annex Building is a L86'-4" x W66'-0" x H55'-0" structure with 2'-0" reinforced concrete exterior walls. Additional protection for this stairwell is not required. The existing Auxiliary Building structure and the adjacent Annex Building structure provides adequate protection against external missiles.

Fire area 1202 AF 05 is also called Stairwell S05. It has modular steel stairs enclosed by concrete structural walls on all four sides. The stairwell is adjacent to the 2'-0" thick reinforced concrete structural Wall L. The other three walls of this stairwell are 1'-0" thick reinforced concrete. It is located in the center of Area 2 of the Auxiliary Building, therefore it is 27'-0" from the 2'-0" thick external Wall 11. The Turbine Building is located adjacent to the Auxiliary Building Wall 11. Stairwell S01 (1202 AF 01) provides an alternate to Stairwell S05. S01 is the normal pathway to and from Main Control Room (Level 4). Additional protection for this stairwell is not required. The existing Auxiliary Building structure and the adjacent Turbine Building and Annex Building structure provides adequate protection against external missiles.

Fire area 1204 AF 02 is also called Stairwell S03. Stairwell S03 is located adjacent to the Shield Building. Therefore one of its enclosure walls is the 3'-0" thick reinforced concrete Shield Building structure. From El. 135'-3" to the Auxiliary Building roof slab (El. 163'-0") the other three walls of this enclosure are non-masonry. Above the roof slab to El. 185'-0" the three stairwell enclosure walls are 2'-0" thick reinforced concrete. From El. 185'-0" to the top of the stair tower at El. 282'-8" the three walls of the stair tower enclosure are steel siding on a structural steel frame which is attached to the Shield Building. There are no adjacent structures above El. 185'-0" that would provide additional protection for this stair tower. However, below El. 185'-0" the Annex Building, which is located to the east of Auxiliary Building would provide additional protection for this area of the Auxiliary Building. Additional protection for this stairwell/stair tower is not required. Emergency access to the PCS valve room would not be necessary in the event that this stair tower were to become inaccessible. Access to safety related equipment by 1204 AF 02 (S03) is not required for safe shutdown.

Fire area 1205 AF 01 is also called Stairwell S04. It has modular steel stairs enclosed by concrete structural walls on two sides and non-masonry walls on the other two sides. The concrete walls are 2'-0" thick above grade and 3'-0" thick below grade. The inside walls that separate the stairwell from the adjacent corridors and the common wall with the adjacent elevator shaft are non-masonry walls. The Annex Building, which is located adjacent to the Auxiliary Building structural Wall I, provides an additional layer of protection against external missiles for the three levels of stairwell S04 that are located above grade. The first two bays of the Annex Building, which are adjacent to this Auxiliary Building stairwell, have 2'-0" thick x 58'-0" H x 66'-0" L concrete walls that are perpendicular to Auxiliary Building Wall I on Annex Building column lines 2 and 4. These two concrete walls are 44'-0" apart. In addition, the floor slab at El. 158 and El. 135'-3" are reinforced concrete slabs supported by steel framing. Additional protection for this stairwell is not required. The existing Auxiliary Building structure and the adjacent structure provides adequate protection against external missiles.

Fire area 1270 AF 12701 is also called Stairwell 06. The Shield Building roof serves as the roof of Stairwell S06. The basic enclosure of the stairwell is a steel frame structure with steel siding. There are no adjacent structures that would provide additional protection for this stairwell. Additional protection for this stairwell is not required. Emergency access to the PCS valve room would not be necessary in the event that these stairs were to become inaccessible. Access to safety related equipment by this stairwell are not required for safe shutdown.

Fire areas 2000 AF 02, 2009 AF 01, 2003 AF 02, 4001 AF 01, 4001 AF 01, 4002 AF 02, and 4003 AF 02 are in buildings without any safety related equipment. Additional protection is not required.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 280.002

Question:

Section 9.5.1.2.1.1 states that the insulating and jacketing material for electrical cables meet the fire and flame test requirements of Institute for Electrical and Electronic Engineers Standard 1202 (IEEE 1202) or IEEE 383 excluding the option to use "flame source, oil or burlap." This statement is not clear as the use of the "alternative" flame source allowed in Section 2.5.4.5 of IEEE 383 is not duplicated in IEEE 1202. During the review of the AP600 the staff approved the use of the 10-inch wide ribbon burner specified in IEEE 383 and IEEE 1202 as the only acceptable cable fire test procedure. Verify that the intent of the statement is solely to exclude the alternative flame source (i.e. oil soaked burlap) for use in the AP1000 as indicated in Item 96 of Table 9.5.1-1.

Westinghouse Response:

The intent is to use only the 10-inch wide ribbon burner cable fire test procedure specified in IEEE 383 and IEEE 1202. No revision is required to the DCD.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 280.004

Question:

Items 75 and 76 of Table 9.5.1-1 state that alternative or dedicated shutdown capability is not necessary. These statements are incorrect and conflict with Item 25 in the same table. As stated in NUREG 1512, Section 9.5.1.1.d, the staff concluded that the safety-related passive core cooling system (PXS) and passive containment cooling system (PCS) used to achieve and maintain safe shutdown following a fire in the AP600 are acceptable as an alternative/dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP. Please correct the discrepancy to be consistent with NUREG 1512 and Item 25 of Table 9.5.1-1 in the AP1000 DCD.

Westinghouse Response:

The DCD will be corrected as shown.

Design Control Document (DCD) Revision:

Correct spelling error in Item 25 of Table 9.5.1-1:

25. Alternative or dedicated shutdown capability should be provided where the protection of systems whose functions are required for safe shutdown is not provided by established fire suppression methods or by Position C.5.b.	C.1.d	AC	In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G.2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutudown systems have not been protected in accordance with the guidance prescribed in the BTP.

Revise Table 9.5.1-1, Item 75:

75. Provision of alternative or dedicated shutdown capability in certain fire areas.	C.5.b (3)	AC	**In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.**~~Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability, as defined in 10 CFR 50 Appendix R, is not necessary.~~

Revise Table 9.5.1-1, Item 76:

76. Alternative or dedicated shutdown capability.	C.5.c	NC	**In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. The safety-related PXS and PCS are used to achieve**

and maintain safe shutdown following a fire and are acceptable as an alternative/ dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.~~Safe shutdown systems are protected such that reliance on alternative or dedicated shutdown capability, as defined in 10 CFR 50 Appendix R, is not necessary.~~

The criteria concerning cold shutdown capability deviates from the criteria applied to the evolutionary reactor designs, but is consistent with the criteria applicable to existing plants. To enhance the survivability of the normal safe shutdown and cold shutdown capability in the event of a fire, and to reduce the reliance on the infrequently utilized safety-related passive systems, automatic suppression is provided in those fire areas outside containment where a fire could damage the normal shutdown capability, or result in a spurious operation of equipment that could result in a venting of the RCS. This criterion does not ensure that the normal shutdown capability will be free of fire damage, or that the equipment necessary to achieve and maintain cold shutdown can be repaired within 72 hours.

PRA Revision:

None

RAI Number: 281.002

Question:

High concentrations of halogens and sulfates present in the system can accelerate the corrosion of components in the DTS. Please provide the maximum allowable concentrations of halogens and sulfates present in the system. (Section 9.2.3)

Westinghouse Response:

The range of halogens and sulfates present in the system are identified in DCD Table 9.2.3-1, the applicable portion of which appears below.

Parameters	Normal Value	Initiate Action
Chloride, ppb	≤ 1	
Sulfate, ppb	≤ 1	

The maximum value is 1 ppb chloride and 1 ppb sulfate.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 281.003

Question:

Please provide the bases for the values governing the blowdown flowrates, i.e., the minimum and maximum percentage values of the maximum steaming rate. (Section 10.4.8)

Westinghouse Response:

Re: DCD section 10.4.8.2.2.2 Normal Operation

The AP600 and AP1000 blowdown system hardware are essentially identical to each other. Thus, on a percentage of maximum steaming rate basis, the AP1000 blowdown system is smaller by a ratio of 8,444,439 to 14,975,742. (Refer to main steam flowrates identified the heat balance figures, 10.1-1, of the AP600 and AP1000 DCDs.) Thus, the minimum and maximum percentage flowrates identified in the AP1000 DCD are similarly reduced compared to the values identified in the AP600 DCD. Specifically, the minimum value was calculated by 0.1% * x 8,444,439 / 14,975,742 = 0.056% => "about 0.06%" and the maximum value was calculated by 1.0% * x 8,444,439 / 14,975,742 = 0.56% => "about 0.6%".

For both the AP600 and the AP1000, during normal operation, the expected flowrate will be approximately 0.1% of the maximum steaming rate on a gpm basis. Thus, the gpm values identified in the AP1000 DCD are approximately 14,975,742 / 8,444,439 x the gpm flowrates identified in the AP600 DCD.

The AP1000 system capacity is sufficient for all anticipated operational needs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.020

Question:

Section 9.1.3.5, "Safety Evaluation," states that safety-related makeup water can be supplied to the SFP from the fuel transfer canal, cask washdown pit, and passive containment cooling water storage tank (PCCWST). Since the passive containment cooling water storage tank is non-safety-related, please explain the circumstances that permit safety-related makeup water to be provided from a non-safety-related source.

Westinghouse Response:

The passive containment cooling water storage tank (PCCWST) is a safety-related tank integral to the top of the shield building. This tank supplies water to the outside of the containment surface to provide safety-related containment cooling. When the decay heat level in the reactor is less than 9 MW (see DCD section 9.1.3.4.3) this water is not needed for containment cooling and can be used for makeup to the spent fuel pool. The piping and valves connecting the PCCWST to the spent fuel pool are safety-related.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.012

Question:

420.12 (DCD 7.1.7, item 7)

DCD 7.1.7, item 7, WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," Section 4.10.2, states that the protection and monitoring system (PMS) provides both system-level and component-level manual means of actuating ESF functions, and the diverse actuation system (DAS) provides manual means of actuating selected ESF functions. To support manual ESF actuation, both the PMS and the DAS provide plant information to the operator. Identify all the PMS and the DAS system-level and component-level manual actuation devices for every ESF function and the related supporting indications to the operator.

Westinghouse Response:

The manually actuated functions of the Protection and Safety Monitoring System (PMS) are described in DCD Tier 1 Section 2.5.2 and DCD Tier 2 Sections 7.1.2 and 7.3. The operator can perform the manual operations as specified in DCD Tables 7.2-4 and 7.3-3 using non-safety soft controls available at the operator workstations. In addition, the PMS provides a minimum inventory of safety-related dedicated manual actuation controls for the system-level actuations as specified in Table 18.12.2-1. The PMS manual controls are located on the Dedicated Safety Panel in the Main Control Room.

DCD Table 7.5-1 provides a list of the post-accident monitoring variables that are provided to the operator. All the variables listed in Table 7.5-1 are available on the non-safety operator workstations. Selected variables, as indicated in Table 7.5-1, are also available on the safety-related Qualified Data Processing Subsystem (QDPS) subsystem of PMS. The QDPS is described in DCD Tier 2 Section 7.1.2.5. Status of the PMS-actuated equipment is also displayed by the QDPS. The minimum inventory of displays and alerts is listed in DCD Tier 1 Table 2.5.2-5 and DCD Tier 2 Table 18.12.2-1.

The manually actuated functions and supporting indications of the Diverse Actuation System (DAS) are described in DCD Tier 1 Section 2.5.1 and DCD Tier 2 Section 7.7.1.11. Dedicated manual actuation controls are provided for the functions specified in DCD Tier 1 Table 2.5.1-2. DAS sensors and displays are listed in DCD Tier 1 Table 2.5.1-3 and described in DCD Tier 2 Section 7.7.1.11. The use of these indications to support diverse manual actuation is described in Section 7.7.1.11 under "Indication" on page 7.7-17. Status of the DAS-actuated equipment is not displayed by DAS, but rather by the post-accident monitoring system variables on QDPS; see Table 7.5-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 420.014

Question:

420.14 (DCD 7.1.7, item 7)

WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," has not addressed specific compliance with Branch Technical Position HICB-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems," in SRP Chapter 7. Discuss the AP1000's compliance with the four point positions listed in the BTP HICB-19.

Westinghouse Response:

The AP1000 evaluation of Defense-in-Depth and Diversity was performed using the same techniques used for AP600.

The primary factors for defense against common-mode failures are quality and diversity in the I&C design. Maintaining high quality increases the reliability of both individual components and complete systems. Diversity in assigned functions (for both equipment and human activities), equipment, hardware, and software reduces the probability of a common-mode failure.

The AP1000, like the AP600, uses passive safety systems that rely on natural forces such as density differences, gravity, and stored energy to provide water for core and containment cooling. The active systems of the AP1000 are not classified as safety-related, and credit is not taken for these active systems in the licensing design-basis analyses described in DCD Chapter 15, unless their operation makes the consequences of an accident more limiting. The non-safety-related active systems in the AP1000 provide defense-in-depth functions and supplement the capability of the safety-related passive systems.

AP1000 compliance with BTP HICB-19 is addressed in WCAP-15775. AP1000 compliance with the four point positions of BTP HICB-19 is summarized below:

POINT 1 – *"The applicant/licensee should assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed."*

AP1000 response: The defense-in-depth and diversity of the proposed instrumentation and control system have been assessed. Vulnerabilities to common-mode failures were addressed and described in WCAP-15775.

POINT 2 – *"In performing the assessment, the vendor or applicant/licensee shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant/licensee shall demonstrate adequate diversity within the design for each of these events."*

AP1000 response: The analysis of common-mode failure in the AP1000 I&C architecture was done as part of the probabilistic risk assessment (PRA). The AP1000 PRA includes an assessment of the spectrum of design basis and beyond design basis events. In the PRA, failures of the instrumentation and control architecture, including common-mode failures, were analyzed. The PRA report describes this analysis of the AP1000 instrumentation and control systems. Chapter 26 of the PRA report, "Protection and Safety Monitoring System," describes the Protection and Safety Monitoring System (PMS) model. Chapter 27 of the PRA report, "Diverse Actuation System," describes the diverse actuation system (DAS) model. The AP1000 PRA results demonstrate adequate diversity within the design for the spectrum of design basis and beyond design basis events.

The PMS is designed to prevent common mode failures. However, in the low probability case where a common mode failure does occur, the DAS provides diverse protection. The specific functions performed by the DAS are selected based on the PRA evaluation. The DAS functional requirements are based on an assessment of the protection system instrumentation common-mode failure probabilities combined with the event probability.

POINT 3 – *"If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, should be required to perform either the same function of a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions."*

AP1000 response: The DAS is a nonsafety-related system that provides a diverse backup to the PMS. The DAS provides an alternate means of initiating reactor trip and actuating selected engineered safety features, and providing plant information to the operator. The DAS is described in DCD subsection 7.7.1.11. The DAS is included to support the aggressive AP1000 risk goals by reducing the probability of a severe accident that potentially results from the unlikely coincidence of postulated transients and postulated common mode failure in the protection and control systems.

The automatic actuation signals provided by the DAS are generated in a functionally diverse manner from the PMS actuation signals. The common-mode failure of sensors of a similar design is also considered in the selection of these functions. Diversity is achieved by the use of a different architecture, different hardware implementations and different software from that of the PMS. Software diversity is achieved by running different operating systems and programming in different languages. Diversity of the DAS is assured by DCD Tier 1 Section 2.5.1, ITAAC 3.

POINT 4 – *"A set of displays and controls located in the main control room should be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls should be independent and diverse from the safety computer systems identified in items 1 and 3 above."*

AP1000 response: Both the PMS and DAS provide manual means of tripping the reactor. The PMS provides a hardwired reactor trip to the reactor trip breakers. The DAS provides a diverse hardwired reactor trip to the rod drive motor-generator set breakers. To support manual reactor trip, both the PMS and the DAS provide plant information to the operator. The PMS provides the Class 1E QDPS indications, while the DAS provides non-safety diverse indications.

The PMS provides both system level and component level manual means of actuating ESF functions. The DAS provides manual means of actuating selected ESF functions. The manual actuation function of the DAS is implemented by wiring the controls located in the main control room directly to the final loads in a way that completely bypasses the normal path through the control room multiplexers, the PMS cabinets and the DAS automatic logic. To support manual ESF actuation, both the PMS and the DAS provide plant information to the operator. The PMS provides the Class 1E QDPS indications, while the DAS provides nonsafety diverse indications.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.010

Question:

For the AP1000 14-foot core design, the 14-foot fuel assemblies tend to have more vibrational problems than the 12-foot fuel assemblies in the currently operating PWRs.

Provide operational experiences, test programs, and/or surveillance plans that demonstrate the flow vibration will not present a serious operational problem to the AP1000 core.

Westinghouse Response:

Contrary to the assertion, Westinghouse experience with 14-foot fuel assemblies is that they tend to have fewer vibration problems than 12-foot assemblies. Among other characteristics, fuel assembly & fuel rod vibration depend on the grid geometry and the span length between grids. The span length between grids for 14-foot fuel is shorter than that in 12-foot fuel. Shorter span lengths tend to improve the vibration performance of the assembly. The AP1000 fuel span length is even shorter than earlier 12-foot and 14-foot designs due to the addition of IFM grids. The grid geometry selected for the AP1000 design, the Robust Fuel Assembly (RFA) grid, is one that has had successful performance in both 12-foot and 14-foot Westinghouse cores.

The RFA mid grid design was developed following extensive fuel assembly and fuel rod vibration testing and the incorporation of field experience of various grid designs. To date it has been introduced in 33 regions of 12-foot fuel and 11 regions of 14-foot fuel. A detailed post irradiation examination, which included fuel rod removal, has been performed at one site and confirmed satisfactory RFA grid vibration performance. Leaking RFA fuel has not been detected in any region except for two debris related occurrences reported by a licensee. Many additional regions will be introduced between now and the first AP1000 core. The AP1000 fuel design will have the benefit of RFA field experience for both 12-foot and 14-foot cores.

During the reactor operations, data from Nuclear Instrumentation System (NIS - excore and incore detectors) and core exit thermocouples can be periodically assessed to insure no unacceptable flow induced vibrations and flow anomaly are presented.

Surveillance programs conducted by operating plants typically include inspection of post-irradiated fuel assemblies. These surveillance programs establish the schedule, guidelines, and inspection criteria for conducting visual inspection of post-irradiated fuel assemblies. They include quantitative visual examination of some discharged fuel assemblies from each refueling.

The programs also include criteria for additional inspection requirements for post-irradiated fuel assemblies if unusual characteristics are noticed in the visual inspection or if plant instrumentation and subsequent laboratory analysis indicates gross failed fuel. The post-irradiated fuel surveillance programs address disposition of fuel assemblies receiving an unsatisfactory visual inspection.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.015

Question:

The 2nd paragraph of Section 4.3.2.3.2.1, "Moderator Density and Temperature Coefficients," makes reference to the possibility of a positive net value for the moderator coefficient. The third paragraph of this same Section states that the value of moderator temperature coefficient (MTC) over the range of power operation is negative.

A. Provide values of the MTC in tabular form or graphical form for values of the moderator coefficients at Hot Zero Power (HZP) and at Hot Full Power (HFP) as a function of core life for an expected typical cycle of a AP1000 core.

B. If indeed the MTC is positive, how will it impact the Technical Specification associated with the MTC?

C. If the MTC is positive, how does it impact the ATWS event with regards to exceeding the vessel pressure of 3200 psia and the amount of time the core is in an "unfavorable Exposure Time (UET)"?

D. The UET domain is a domain where the vessel pressure is known to be exceeded. Please provide the range of vessel pressures for the total time the core is in this domain, or as a function of core life for an expected typical cycle of a AP1000 core.

Westinghouse Response:

The AP1000 moderator temperature coefficient is negative when the reactor is critical. During power operation, the moderator temperature coefficient is sufficiently negative to support ATWT rideout capability over the entire equilibrium cycle and at any power level with the reactor coolant at the normal operating temperature. During shutdown or refueling conditions, when boron concentrations are high, the MTC can be positive (see DCD Figure 4.3-21).

A. Table 440.015-1 provides the AP1000 MTC versus Cycle Burnup for Hot Full Power (HFP) and Hot Zero Power (HZP), no Xenon. The Hot Full Power Temperature Coefficient versus Cycle Burnup information provided in Table 440.015-1 is provided as AP1000 DCD Figure 4.3-24.

B. The AP1000 moderator temperature coefficient is not positive during power operation.

C. The AP1000 moderator temperature coefficient is not positive during power operation.

D. The AP1000 ATWS analysis presented in Appendix A of the AP1000 PRA demonstrates that the peak AP1000 RCS pressure is less than 3200 psig with a UET of 0. This is a result of the operation of the passive safety systems, as well as the lower MTC associated with the lower core boron concentration of the AP1000. Refer to PRA Appendix A for additional discussion of the AP1000 ATWS analysis assumptions. In addition, see RAI 440.014 for additional information regarding the compliance of the AP1000 ATWS analysis to the ATWS Rule.

Table 440.015-1

AP1000 Cycle 1 – Moderator Temperature Coefficient vs Cycle Burnup

Burnup (MWD/MTU)	HFP MTC (1) (pcm/°F)	HZP MTC (pcm/°F)
0	-10.357	-5.250
150	-13.165	-5.151
1000	-11.925	-3.498
2000	-10.572	-1.877
3000	-10.343	-1.378
4000	-10.461	-1.196
5000	-11.084	-1.524
6000	-11.864	-2.022
8000	-13.697	-3.182
10000	-16.004	-4.853
12000	-18.392	-6.656
14000	-21.049	-8.805
16000	-24.022	-11.159
18000	-26.719	-13.629
20000	-29.709	-16.080
20880	-31.247	-17.169

Notes:
(1) Information provided in AP1000 DCD Figure 4.3-24.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.020

Question:

Section 4.3.2.3 states that the reactivity coefficients are calculated with approved nuclear methods. Please provide reference to these approved methods.

Westinghouse Response:

DCD Section 4.3.5, Reference 40 and Reference 57 address reactivity coefficients and predictive capability of the PHOENIX-P/ANC Nuclear Design System.

From DCD Section 4.3.5:

40. Nguyen, T. Q., et. al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, (Proprietary) June 1988

57. Davidson, S. L., (Ed.), et. al., "ANC: Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, (Proprietary) September 1986.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.021

Question:

Tier 2 Information Section 4.4.4.5.2 cites WCAP-14565-P-A for the NRC approval of the VIPRE-W core model, and references (Reference 83) WCAP-14565-P-A with the title of "VIPRE-W Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis." However, the formal title of WCAP-14565-P-A topical report has the word "VIPRE-01" instead of "VIPRE-W."

Confirm that the VIPRE-W code is the same version of VIPRE-01 code discussed in the WCAP-14565-P-A topical report. Discuss whether Westinghouse intends to change the word in the WCAP-14565-P-A topical report from "VIPRE-01" to "VIPRE-W

Westinghouse Response:

The VIPRE-W code is the same as VIPRE-01 discussed in the WCAP-14565-P-A topical report. Westinghouse does not intend to change the word in WCAP-14565-P-A from VIPRE-01 to VIPRE-W. The AP1000 DCD will be modified, as indicated below, to change VIPRE-W to VIPRE-01.

Design Control Document (DCD) Revision:

From DCD Section 1.6, Table 1.6-1 (Sheet 10 of 20):

WCAP-14565-P-A (P) WCAP-15306-NP-A	~~VIPRE-W~~ VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999

From DCD Section 4.1, Table 4.1-2 (Sheet 2 of 2):

Thermal-hydraulic design steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	~~VIPRE-W~~ VIPRE-01	4.4.4.5.2

Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel.	~~VIPRE-W~~ VIPRE-01	4.4.4.5.4

From DCD Section 4.4.1.1.2:

For those transients that use the ~~VIPRE-W~~ VIPRE-01 computer program (subsection 4.4.4.5.2) and the WRB-2M correlation (subsection 4.4.2.2.1), the Revised Thermal Design Procedure design limits are 1.25 for the typical cell and 1.25 for the thimble cell for Core and Axial Offset Limits and 1.22 for the typical cell and 1.21 for the thimble cell for all other RTDP transients. These values may be revised (slightly) when plant specific uncertainties are available.

From DCD Section 4.4.2.2:

DNBRs are calculated by using the correlation and definitions described in subsections 4.4.2.2.1 and 4.4.2.2.2. The ~~VIPRE-W~~ VIPRE-01 computer code described in subsection 4.4.4.5, is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is described in subsections 4.4.4.3.1 (nuclear hot channel factors) and 4.4.2.2.4 (engineering hot channel factors).

Sections From DCD 4.4.2.2.3:

The application of the thermal diffusion coefficient in the ~~VIPRE-W~~ VIPRE-01 analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 83.

The data from these tests on the R-mixing vane grid show that a design thermal diffusion coefficient value of 0.038 (for 26-inch grid spacing) can be used in determining the effect of coolant mixing in the THINC analysis. An equivalent value of the mixing coefficient is used in the ~~VIPRE-W~~ VIPRE-01 evaluations (Reference 83). A mixing test program similar to the one just described was conducted for the current 17 x 17 geometry and mixing vane grids on 26-inch spacing, as described in WCAP-8298-P-A (Reference 16). The mean value of the thermal diffusion coefficient obtained from these tests is 0.059.

From DCD Section 4.4.2.2.4:

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise is directly considered in the ~~VIPRE-W~~ VIPRE-01 core thermal subchannel analysis, described in subsection 4.4.4.5.1 under any reactor opening condition. The following items are considered as contributors to the enthalpy rise engineering hot channel factor:

- Pellet diameter, density, and enrichment

Variations in pellet diameter, density, and enrichment are considered statistically in establishing the limit DNBRs, described in subsection 4.4.1.1.2, for the Revised Thermal Design Procedure (Reference 2). Uncertainties in these variables are determined from sampling of manufacturing data.

- Inlet flow maldistribution

  The consideration of inlet flow maldistribution in core thermal performances is described in subsection 4.4.4.2.2. A design basis of five-percent reduction in coolant flow to the hot assembly is used in the ~~VIPRE-W~~ VIPRE-01 analyses.

- Flow redistribution

  The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the ~~VIPRE-W~~ VIPRE-01 analyses for every operating condition evaluated.

- Flow mixing

The subchannel mixing model incorporated in the ~~VIPRE-W~~ VIPRE-01 code and used in reactor design is based on experimental data, as detailed in WCAP-7667-P-A (Reference 18) and discussed in subsections 4.4.2.2.3 and 4.4.4.5.1. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The ~~VIPRE-W~~ VIPRE-01 mixing model is discussed in Reference 83.

From DCD Section 4.4.2.5:

The ~~VIPRE-W~~ VIPRE-01 code is used to calculate the flow and enthalpy distribution in the core for use in safety analysis. Extensive experimental verification of ~~VIPRE-W~~ VIPRE-01 is presented in Reference 84

From DCD Section 4.4.2.7.3:

~~VIPRE-W~~ VIPRE-01 considers two-phase flow in two steps. First, a quality model is used to compute the flowing vapor mass fraction (true quality) including the effects of subcooled boiling. Then, given the true void quality, a bulk void model is applied to compute the vapor volume fraction (void fraction).

~~VIPRE-W~~ VIPRE-01 uses a profile fit model (Reference 83) for determining subcooled quality. It calculates the local vapor volumetric fraction in forced convection boiling by: 1) predicting the point of bubble departure from the heated surface and 2) postulating a relationship between the true local vapor fraction and the corresponding thermal equilibrium value.

From DCD Section 4.4.2.9.5:

The uncertainties in the DNBRs calculated by the ~~VIPRE-W~~ VIPRE-01 analyses, discussed in subsection 4.4.4.5.1, due to uncertainties in the nuclear peaking factors are accounted for by applying conservatively high values of the nuclear peaking factors. Measurement error allowances are included in the statistical evaluation of the limit DNBR described in subsection 4.4.1.1 using the Revised Thermal Design Procedure. More information is provided in WCAP-11397-P-A (Reference 2). In addition, conservative values for the engineering hot channel factors are used as presented in subsection 4.4.2.2.4. The results of a sensitivity study, WCAP-8054-P-A (Reference 22), with THINC-IV, a ~~VIPRE-W~~ VIPRE-01 equivalent code, show that the minimum DNBR in the hot channel is relatively insensitive to variations in the core-wide radial power distribution (for the same value of $F^N_{\Delta H}$).

The ability of the ~~VIPRE-W~~ VIPRE-01 computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in subsection 4.4.4.5.1 and in Reference 83. Studies (Reference 84) have been performed to determine the sensitivity of the minimum DNBR to the void fraction correlation (see also subsection 4.4.2.7.3) and the inlet flow distributions. The results of these studies show that the minimum DNBR is relatively insensitive to variation in these parameters. Furthermore, the ~~VIPRE-W~~ VIPRE-01 flow field model for predicting conditions in the hot channels is consistent with that used in the derivation of the DNB correlation limits including void/quality modeling, turbulent mixing and crossflow and two phase flow (Reference 83).

From DCD Section 4.4.2.9.8:

4.4.2.9.8 Uncertainty in Mixing Coefficient

A conservative value of the mixing coefficient, that is, the thermal diffusion coefficient, is used in the ~~VIPRE-W~~ VIPRE-01 analyses.

From DCD Section 4.4.4.2.2:

A core inlet flow distribution reduction of five percent to the hot assembly inlet is used in the ~~VIPRE-W~~ VIPRE-01 analyses of DNBR in the AP1000 core. Studies shown in WCAP-8054-P-A (Reference 22), made with THINC-IV, a ~~VIPRE-W~~ VIPRE-01 equivalent code, show that flow distributions significantly more nonuniform than five percent have a very small effect on DNBR, which is accounted for in the DNB analysis.

Sections from DCD 4.4.4.2.3:

The friction factor for ~~VIPRE-W~~ VIPRE-01 in the axial direction, parallel to the fuel rod axis, is evaluated using a correlation for a smooth tube (Reference 83). The effect of two-phase flow on the friction loss is expressed in terms of the single-phase friction pressure drop and a two-phase friction multiplier. The multiplier is calculated using the homogenous equilibrium flow model.

The comparisons of predictions to data given in Reference 83 verify the applicability of the ~~VIPRE-W~~ VIPRE-01 correlations in PWR design.

From DCD Section 4.4.4.5.1:

The objective of reactor core thermal design is to determine the maximum heat removal capability in all flow subchannels and to show that the core safety limits, as presented in the technical specifications, are not exceeded while combining engineering and nuclear effects. The thermal design takes into account local variations in dimensions, power generation, flow redistribution, and mixing. ~~VIPRE-W is the~~ The Westinghouse version of

VIPRE-01, a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels, is described in Reference 83 ~~(Reference 84)~~,. ~~VIPRE-W~~ VIPRE-01 modeling of a PWR core is based on a one-pass modeling approach (Reference 83). In the one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon the inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow and pressure drop.

From DCD Section 4.4.4.5.2:

The ~~VIPRE-W~~ VIPRE-01 core model as approved by the NRC (Reference 83) is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions. The VIPRE-01 code is described in detail in Reference 84, including discussions on code validation with experimental data. The ~~VIPRE-W~~ VIPRE-01 modeling method is described in Reference 83, including empirical models and correlations used. The effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the ~~VIPRE-W~~ VIPRE-01 evaluations. However, conservative treatment by the ~~VIPRE-W~~ Westinghouse VIPRE-01 modeling method has been demonstrated to bound this effect in DNBR calculations (Reference 83).

From DCD Section 4.4.4.5.3:

Extensive additional experimental verification of VIPRE-01 is presented in Reference 84.

The ~~VIPRE-W~~ VIPRE-01 analysis is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. The use of the ~~VIPRE-W~~ VIPRE-01 analysis provides a realistic evaluation of the core performance and is used in the thermal hydraulic analyses as described above.

From DCD Section 4.4.4.5.4:

~~VIPRE-W~~ VIPRE-01 is capable of transient DNB analysis. The conservation equations in the ~~VIPRE-W~~ VIPRE-01 code contain the necessary accumulation terms for transient calculations. The input description can include one or more of the following time dependent arrays:

From DCD Section 4.4.4.7:

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior are more pronounced than external blockages of the same magnitude. In both cases, the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where the reduction occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the ~~VIPRE-W~~ VIPRE-01 program. Inspection of the DNB correlation (subsection 4.4.2.2 and References 4, 5, and 6) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The ~~VIPRE-W~~ VIPRE-01 code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. Reference 84 shows that, for a fuel assembly similar to the Westinghouse design, ~~VIPRE-W~~ VIPRE-01 accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reactor operating at the nominal full-power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the fuel rods reaching the DNBR limit.

From DCD Section 4.4.5.2:

Additional demonstration of the overall conservatism of the THINC analysis was obtained by comparing THINC predictions to in-core thermocouple measurements, as detailed WCAP-8453-A (Reference 78). ~~VIPRE-W~~ VIPRE-01 has been confirmed to be as conservative as the THINC code in Reference 83.

From DCD Section 4.4.8:

83. Sung, Y. X., et al., "~~VIPRE-W~~ VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A and WCAP-15306-NP-A, October 1999.

From DCD Section 4.4, Table 4.4-2:

Table 4.4-2

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS WITH DESIGN HOT CHANNEL FACTORS

(BASED ON ~~VIPRE-W~~ VIPRE-01)

	Average	Maximum
Core, %	0.0	-
Hot Subchannel, %	0.1	0.9

From DCD Section 15.0.3.3:

The radial and axial power distributions just described are input to the ~~VIPRE~~ VIPRE-01 code as described in subsection 4.4.

From DCD Section 15.0, Table 15.0.2: See response to RAI 440.055 for revisions to Table 15.0.2.

From DCD Section 15.1.2.2.1:

For that portion of the feedwater malfunction transient that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

From DCD Section 15.1.3.2.1:

For the excessive load increase analysis that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

From DCD Section 15.1.5.2.1:

- The thermal-hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic digital computer code, ~~VIPRE~~ VIPRE-01, is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

From DCD Section 15.1.6.2.1:

Analysis performed for inadvertent PRHR heat exchanger actuation including a primary coolant flow coastdown caused by the consequential loss of offsite power uses a combination of three computer codes for the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate heat flux based on the nuclear power and flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

From DCD Section 15.2.3.2.1:

In the turbine trip analyses, that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the departure from nucleate boiling ratio (DNBR) analyses. First, the LOFTRAN code (References 2 and 6) is used to calculate the plant system transient. The FACTRAN code (Reference 7) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

From DCD Section 15.3.1.2.1:

This transient is analyzed using three computer codes. First, the LOFTRAN code (Reference 1) is used to calculate the core flow during the transient based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients as predicted from the loss of two reactor coolant pumps. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical cell or the thimble cell.

From DCD Section 15.4.1.2.1:

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in ~~VIPRE-W~~ VIPRE-01 (described in Section 4.4) for the transient DNBR calculation.

From DCD Section 15.4.2.2.1:

For that portion of the RCCA withdrawal at-power analysis that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First, the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 2) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the ~~VIPRE-W~~ VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow, inlet core temperature (and pressure) from LOFTRAN.

Sections from DCD 15.4.3.2.1.1:

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system transient conditions and reactor power. By combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling design basis is shown to be met using the ~~VIPRE-W~~ VIPRE-01 code.

- Statically misaligned RCCA

Steady-state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the ~~VIPRE-W~~ VIPRE-01 code to calculate the DNBR.

Sections from DCD 15.4.3.2.2.1:

Power distributions within the core are calculated using the computer codes described in Table 4.1-2. The peaking factors are then used by ~~VIPRE-W~~ VIPRE-01 to calculate the DNBR for the event. The case of the worst rod withdrawn from the mechanical shim or axial offset bank inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at beginning of life because this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Sections from DCD 15.6.1.2.1:

For reactor coolant system depressurization analyses that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analyses. First the LOFTRAN code is used to perform the plant system transient. The FACTRAN code (Reference 18) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the ~~VIPRE~~ VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.022

Question:

Section 4.4.1.1.2 states that for those transients that use the VIPRE-W computer program and the WRB-2M correlation, the Revised Thermal Design Procedure (RTDP) design limits are 1.25 for the typical cell and 1.25 for the thimble cell for Core and Axial Offset Limits, and 1.22 for the typical cell and 1.21 for the thimble cell for all other RTDP transients, and that these values may be revised when plant specific uncertainties are available.

A. Discuss the differences between the RTDP design departure from nucleate boiling ratio (DNBR) limits for (1) core and axial offset limits, and (2) other RTDP transients, respectively.

B. Provide the derivations of these RTDP design DNBR limits, including the uncertainties of all parameters used in the derivation.

C. Provide the instrument uncertainty methodology and the assumed uncertainty values of various components of the instrument for the measurements of the parameters included in the RTDP.

Westinghouse Response:

A. RTDP Design Limits are calculated using parameter uncertainties and DNBR sensitivities to these parameters for a number of conditions as illustrated in WCAP-11397-P-A, "Revised Thermal Design Procedure." The magnitude of the DNBR sensitivities to the various parameters is dependent upon the conditions analyzed. The calculations that are associated with Core Limit conditions gave higher DNBR Design Limits than those associated with the other RTDP conditions. To maximize margin, separate DNBR Design Limits are used for each set of conditions.

B. The calculation of the RTDP DNBR Design Limits follows that illustrated in WCAP-11397-P-A, "Revised Thermal Design Procedure." The following values were used for sigma: Power = 1.0%, Tin = 3 degrees F, Pressure = 30 psi, Flow = 1.25%, Bypass = 0.5%, FdH = 0.0386, FdHE1 = 0.0182, VIPRE code = 0.02, Transient code = 0.005. The values for Power, Tin, Pressure and Flow were assumed since the plant instrumentation to measure these has not been detailed. These are typical bounding values. The calculations will be revised when the plant is built. Experience has shown that any changes in these parameters are expected to have a minor impact (less than 1%) on the design limits. The value for FdH is based on a 4% uncertainty and a FdH value equal to 1.587. These DNBR Design Limits were based on the WRB-2M DNB correlation which was based on 241 data points with a mean of 1.0008 and a sample standard deviation of 0.0652.

C. The instrumentation uncertainty methodology will be similar to that used for AP600 in WCAP-14605, "Westinghouse Setpoint Methodology for Protection Systems", April 1996.

The assumed 2-sigma instrumentation uncertainties associated with the sigmas in (B) above are: Power = 2.0%, Tin = 6 degrees F, Pressure = 60 psi and Flow = 2.5%.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.025

Question:

Section 4.4.2.2.1 states that the WRB-2 or W-3 correlation is used wherever the WRB-2M correlation is not applicable; and that the W-3 correlation is used in the heated region below the first mixing vane grid and in the analysis of accident conditions where the system pressure is below the range of the primary correlation.

A. Are both the WRB-2 and W-3 correlations applicable to the AP1000 RFA [robust fuel assembly] XL fuel design?

B. Describe under what conditions will the WRB-2 correlation be used in place of WRB-2M and how is this done in the VIPRE-W code.

C. Do the test assemblies used for the development of the WRB-2 correlation have the same grid design as that of AP1000 RFA XL grid design?

D. Clarify, and explain, if the W-3 correlation is used in the heated region below the first mixing vane for all thermal-hydraulic conditions where the WRB-2M and WRB-2 correlations are applicable, and how this is done in the VIPRE-W code.

E. Since the WRB-2M correlation is used in combination with the RTDP methodology, will the use of the WRB-2 or W-3 correlation in place of WRB-2M be applied in the same fashion? What are the RTDP DNBR limits for Core and Axial Offset limits and for RTDP transients for the WRB-2 and W-3 correlations, respectively? Provide derivations of these limits.

Westinghouse Response:

A. Both the WRB-2 (Reference 1 & 2) and the W-3 DNB correlations are applicable to the AP1000 RFA.

B. The WRB-2 correlation would be used where the conditions were slightly outside of the range of applicability of the WRB-2M correlation. Examples are quality, X and mass flow rate, G:

	WRB-2	WRB-2M
Upper limit on quality	0.3	0.29
Upper limit on flow, lb/hr-ft2	3.7E6	3.1E6

The W-3 correlation would be used, if applicable, if the limits were outside the range of applicability of the WRB-2 or WRB-2M correlations.

The WRB-2 correlation is built into the VIPRE code.

C. The DNB test assemblies for the WRB-2 correlation had a different grid design than the AP1000 RFA XL grid design. However, use of the WRB-2 correlation for the AP1000 RFA XL grid design has been justified in References 1 & 2.

D. The WRB-2M and WRB-2 correlations are based on mixing vane grids and are thus not applicable below the first mixing vane grid. The W-3 correlation is used in the heated region below the first mixing vane grid. All of these correlations are built into the VIPRE code.

E. The RTDP methodology was only applied for conditions where the WRB-2M correlation is applicable. Standard (non-RTDP) methodology was used when the other correlations were used.

References:

1. Letter from N. J. Liparulo (Westinghouse) to J.E. Lyons (NRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications", NSD-NRC-97-5198, June 30, 1997.
2. Letter from H. Sepp (Westinghouse) to T. E. Collins (NRC), "Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design," NSD-NRC-98-5796, October 13, 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.026

Question:

Section 4.4.2.2.4 states that the effects of variations in pellet diameter, density, and enrichment on enthalpy rise engineering hot channel factor are considered statistically in establishing the limit DNBRs for the Revised Thermal Design Procedure, and that uncertainties in these variables are determined from sampling of manufacturing data.

Provide the assumed uncertainty values, and bases, assigned in the RTDP for the variations in pellet diameter, density, and enrichment of the AP1000 RFA XL fuel design.

Westinghouse Response:

The effects of the variations in pellet diameter, density, enrichment (and burnable absorber) are accounted for in the FdHE1 parameter used to calculate the RTDP DNBR design limits. The uncertainty value used was 3%. This is based on measurements from numerous fuel regions.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.041

Question:

AP1000 Technical Specification Bases B 3.4.4 describes that the AP1000 RCPs are powered by variable speed controllers when the reactor coolant temperature is below 500°F, that the pumps must be started using the variable speed controller with the reactor trip breakers open, and that the controller shall be bypassed prior to closure of the reactor trip breakers. There is no discussion of the RCP and controller operation in Tier 2 Section 5.4.1, except that Table 5.4-1 specifies the RCP motor currents for starting, and nominal input, cold reactor coolant.

Provide a detailed description on the design and operation of the variable current RCPs and the variable speed controller, and modify Section 5.4.1, as necessary.

Westinghouse Response:

Each AP1000 canned motor reactor coolant pump is connected to a variable frequency drive which is utilized for pump startup and operation until the reactor coolant system temperature has reached 450 F. Above a reactor coolant temperature of 450 F the variable frequency drives are bypassed and the pumps are run at constant speed. Therefore, for power operations, the AP1000 pumps run at constant speed (same as AP600 pumps).

Each variable frequency drive is an integrated component consisting of transformer, power electronics, output reactor for synchronization, controls and heat exchanger for heat removal. The variable frequency drives are water cooled with plant component cooling water. The drives are designed to limit output harmonics such that no significant changes in the reactor coolant pump design are required for operation with the variable frequency drives.

The variable frequency drives enable the startup of the reactor coolant pumps at slow speeds to decrease the power required from the pump motor during operation at cold conditions. The variable frequency drive provides operational flexibility during pump startup and reactor coolant system heatup. During a plant startup the general startup procedure for the pumps is to start the pumps at a low speed. The speed of the pumps is then ramped to a speed that results in nearly the maximum pump current. This speed is maintained as the reactor coolant system temperature is increased from the heat input from the pumps. As the reactor coolant system temperature increases, the fluid density decreases and the motor current required at the set speed decreases. The operator can then increase the speed to bring the pump motor current back to near the maximum value to maximize the heat input to the reactor coolant system (in order to minimize the heatup time).

Once the reactor coolant system temperature reaches 450 F, the variable frequency controllers are bypassed and the pumps run at constant speed. Reactor coolant system heatup then continues as in current plants until conditions are reached at which power operations can begin. During all power operations (Modes 1, and 2) the variable frequency drives are bypassed and the pumps run at constant speed as in the AP600. Therefore, pump behavior during at-power design basis events is the same as for the constant speed pumps used in the AP600.

DCD sections 5.4.1.2.1 and 5.4.1.2.2 will be modified as shown below.

Design Control Document (DCD) Revision:

From DCD page 5.4-2, section 5.4.1.2.1:

The reactor coolant pump driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. It is designed for removal from the casing for inspection, maintenance and replacement, if required. The stator can protects the stator (windings and insulation) from the controlled portion of the reactor coolant circulating inside the motor and bearing cavity. The can on the rotor isolates the copper rotor bars from the system and minimizes the potential for the copper to plate out in other areas.

The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and through a thermal barrier between the pump casing and the rest of the motor internals. Inside the cooling jacket are coils filled with circulating rotor cavity coolant. This rotor cavity coolant is a controlled volume of reactor coolant that circulates inside the rotor cavity. After the rotor cavity coolant is cooled in the cooling jacket, it enters the lower end of the rotor and passes axially between the rotor and stator cans to remove heat from the rotor and stator.

Each pump motor is driven by a variable speed drive which is utilized for pump startup and operation until the reactor coolant system temperature has reached 450 F. Above a reactor coolant temperature of 450 F the variable frequency drives are bypassed and the pumps run at constant speed.

From DCD page 5.4-3, section 5.4.1.2.2:

5.4.1.2.2 Description of Operation

Reactor coolant is pumped by the main impeller. It is drawn through the eye of the impeller and discharged via the diffuser out through the radial discharge nozzle in the side of the casing. Once the motor housing is filled with coolant, the labyrinth seals around the shaft between the impeller and the thermal barrier minimize the flow of coolant into the motor during operation.

An auxiliary impeller at the lower part of the rotor shaft circulates a controlled volume of the coolant through the motor cooling coils. The coolant is cooled to about 150°F by component cooling water circulating around the cooling coils in the cooling jacket outside the stator shell. The cooled reactor coolant then passes through the annulus between the rotor and stator cans, where it removes heat from the rotor and stator and lubricates the motor's hydrodynamic bearings.

The variable frequency drives enable the startup of the reactor coolant pumps at slow speeds to decrease the power required from the pump motor during operation at cold conditions. The variable frequency drive provides operational flexibility during pump startup and reactor coolant system heatup. During a plant startup, the general startup procedure for the pumps is for the operator to start the pumps at a low speed. During reactor coolant system heatup the pumps are run at the highest speed that is within the allowable motor current limits. As the reactor coolant temperature increases, the allowable pump speed also increases. Once the reactor coolant system temperature reaches 450 F, the variable frequency controllers are bypassed and the pumps run at constant speed.

During all power operations (Modes 1, and 2), the variable frequency drives are bypassed and the pumps run at constant speed.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.043

Question:

Section 5.4.2.1 states that Chapter 15 discusses the accident analysis of a steam generator tube rupture, which is based on a rupture of one tube.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 12, 1993, the NRC staff states its position that an applicant for a passive pressurized-water reactor (PWR) design certification should assess features to mitigate the amount of containment bypass leakage that could result from the rupture of multiple steam generator (SG) tubes. This position arises from a concern that an multiple-tube rupture event creates the likelihood of a SG safety valve (SGSV) lifting and then failing to close, resulting in an unisolable release to the environment bypassing the containment.

A. Discuss the AP1000 design features that mitigate or prevent SGSV challenges during an event of rupture of multiple steam generator tubes.

B. Provide an analysis of multiple-tube rupture events to address the concern of containment bypass leakage resulting from a potential failure of the SGSV to reclose.

Westinghouse Response:

A. The objectives of the AP1000 plant response to multiple steam generator tube rupture are the same as for the AP600, including:

   - To automatically terminate the loss of reactor coolant (RC) without overfilling the steam generator (SG) or opening the secondary safety valves. Operation of the automatic depressurization system (ADS) should not occur during these accidents.
   - In the unlikely event of a secondary safety valve failing open, to provide defense-in-depth core cooling through the use of the ADS and passive safety injection.

   The AP1000 passive systems provide a unique response to the SGTR initiating event with respect to conventional plants by automatically terminating the loss of reactor coolant without depressurizing the reactor coolant system (RCS) or overfilling the SG. The passive residual heat removal (PRHR) heat exchanger acts to reduce the RCS pressure below the pressure of the secondary system and isolate the break flow to the faulted SG. The heat is removed from the RCS through the PRHR instead of the intact SG PORV to stop the leak to the faulted SG. The core makeup tanks (CMTs) provide heat removal and coolant inventory makeup for shrinkage in the RCS. During a SGTR, the CMTs inject water in the recirculation mode, exchanging cold borated water for hot RCS water. The CMTs do not drain during recirculation injection, and therefore, automatic depressurization system (ADS) is not actuated. Low CMT level is the initiation setpoint for ADS.

The AP1000 also provides additional defense-in-depth to mitigate multiple SGTRs. The active, nonsafety related systems can be used to mitigate the multiple SGTR as in a conventional plant, however, in the AP1000 this is not the safety case presented in the SSAR. The intact SG PORV is used to control the RCS pressure and isolate the break. The Chemical and Volume Control System (CVS) auxiliary spray is used to reduce the RCS pressure to allow the pumped residual heat removal system to provide borated makeup flow to the system until the break is isolated. In case of failure of both the active nonsafety related mitigation and also the passive safety related PRHR HX mitigation, the AP1000 provides another defense-in-depth method of mitigation. This method uses the ADS and passive safety injection.

If the secondary system safety valve is arbitrarily assumed to fail open, the PRHR HX will not be able to terminate the loss of RC. The loss of primary system coolant through the SG tube and the stuck open valve eventually causes the CMTs to drain to the ADS actuation setpoint. Actuation of ADS depressurizes the RCS in a controlled, staged manner. The safety related passive injection systems, CMTs, accumulators and IRWST gravity injection provide inventory makeup and boration throughout the depressurization. The core remains covered and cooled through out the sequence, and the plant achieves a safe, stable configuration without a release of fission products from the fuel matrix. Preventing the release of fission products from core mitigates the beyond-design-basis containment bypass.

B. Multiple SG tube ruptures are considered to be beyond design basis accidents. Analyses of two 5-tube multiple-SGTR cases for the AP1000 have been performed with the MAAP4.04 code using best estimate decay heat and best estimate assumptions, except as noted below. The cases are based on the cases presented for the AP600 in reference 1.

B.1 Case 1 – Multiple SGTR with Passive System Response

Case 1 is a passive system mitigation case with PRHR heat operation. The accident is initiated by the simultaneous double-ended failure of five cold side tubes at the top of the tubesheet. Startup feedwater (SFW) and the CVS are conservatively assumed to function because they tend to make the accident worse. The SFW controls operate normally and throttle the startup feedwater based on the nominal SG operating level. The CVS provides RCS makeup until it is isolated on a hi-2 SG narrow range level. The secondary system PORV is assumed to not open.

The MAAP4 results are presented in Figure 440.43-1 through 440.43-9. The results show that the faulted SG does not overfill and the safety valves do not open. Therefore, bypass does not occur.

B.2 Case STK – Multiple SGTR with Failed Open SG Safety Valve

Case STK is a passive system mitigation case with minimum PRHR heat removal. The accident is initiated by the simultaneous double-ended failure of five cold side tubes at the top of the tubesheet. SFW and the CVS are conservatively assumed to function because they tend to make the accident worse. The SFW control is assumed to malfunction such that the SFWS flow continues when the SG level increases above the normal level until it is isolated by a hi-2 SG narrow range level. . The CVS provides RCS makeup until it is isolated on a hi-2 SG narrow range level. The secondary system PORV is assumed to not open. The combination of the low PRHR heat removal and the high SG level control causes the faulted SG pressure to exceed the safety valve setpoint. When the valve opens, it is assumed to fail open although the SG is not predicted to overfill.

The MAAP4 results for case STK are presented in Figure 440.43-10 through 440.43-18. The loss of coolant from the RCS eventually drains the CMTs to the ADS actuation setpoint. The RCS depressurizes and gravity injection begins. The core remains covered and cooled, thus no significant fission product release occurs. Boron dilution due to secondary system water ingress in the RCS during depressurization is not a problem because the PXS injection tanks (CMTs, Accumulators and IRWST) provide boron to the RCS and because boron collects in the secondary system prior to ADS (reference 1). Therefore, the boron concentration of the water coming back into the RCS is expected to have approximately the same boron concentration as the water in the RCS.

References

1. WCAP-14991, AP600 Multiple Steam Generator Tube Rupture Analysis Report.

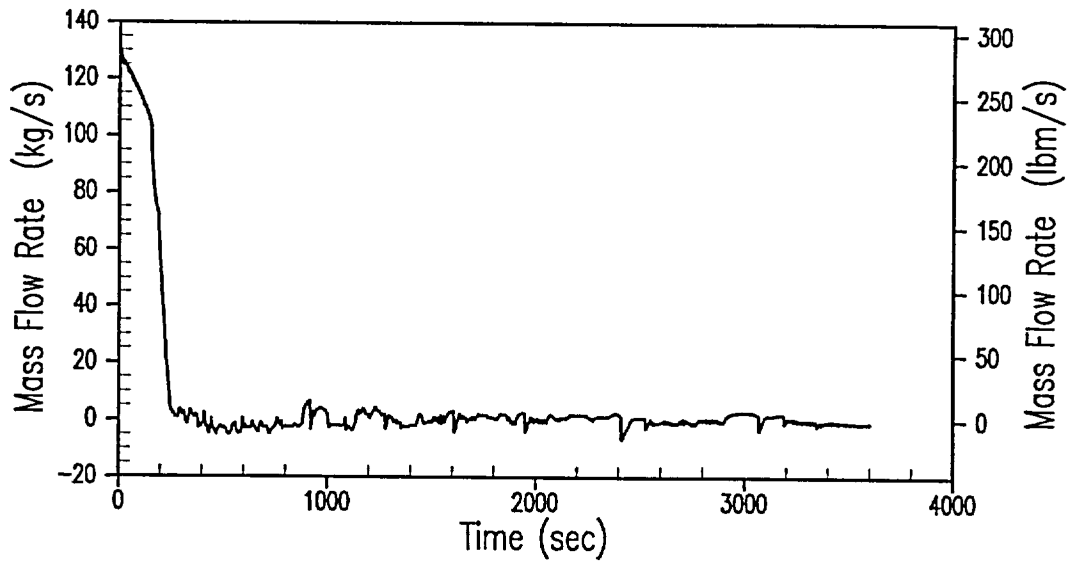


Figure 440.43-1 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Tube Rupture Break Flow

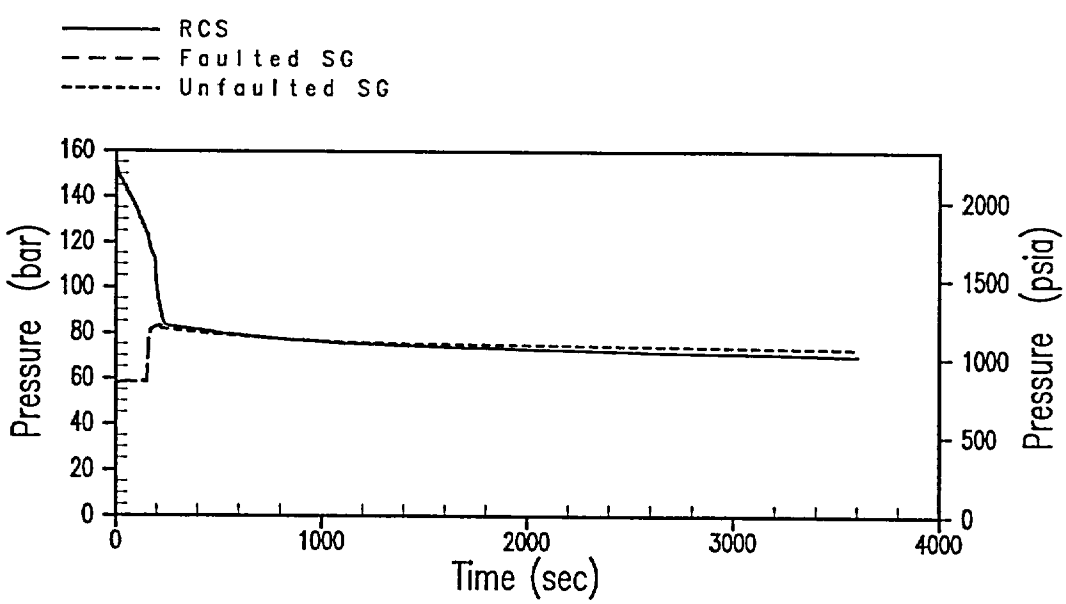


Figure 440.43-2 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS and Secondary System Pressures

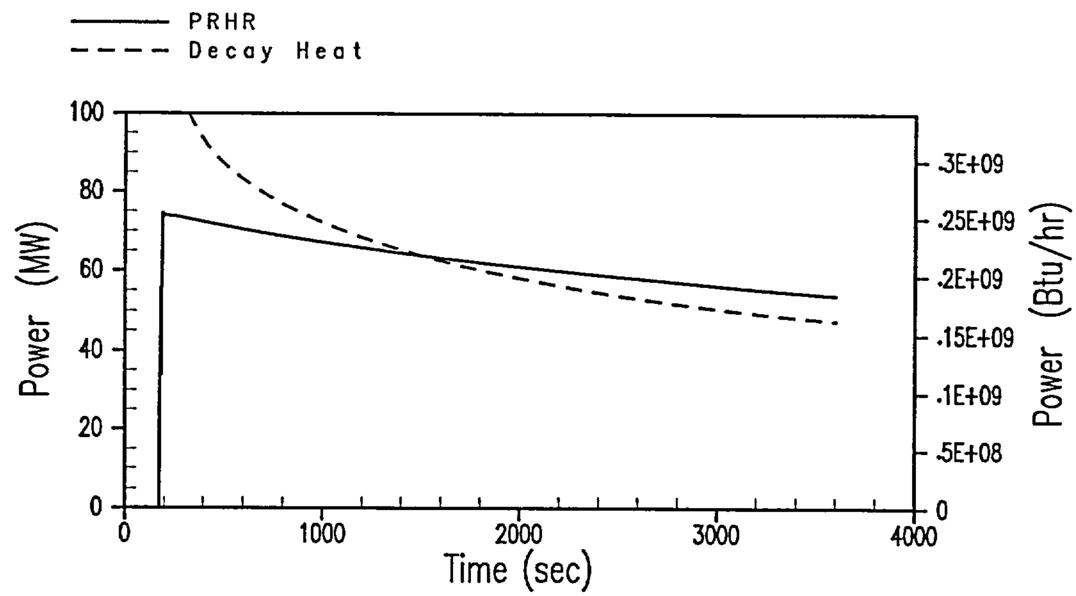


Figure 440.43-3 AP1000 Multiple (5 Tubes) SGTR with PORV Failure PRHR Heat Removal

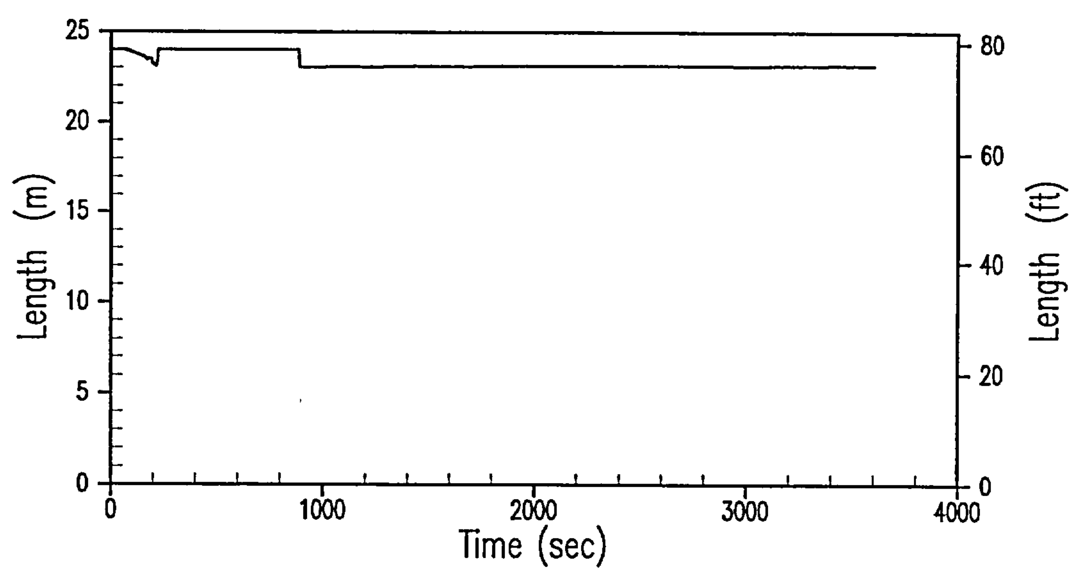


Figure 440.43-4 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS Water Level

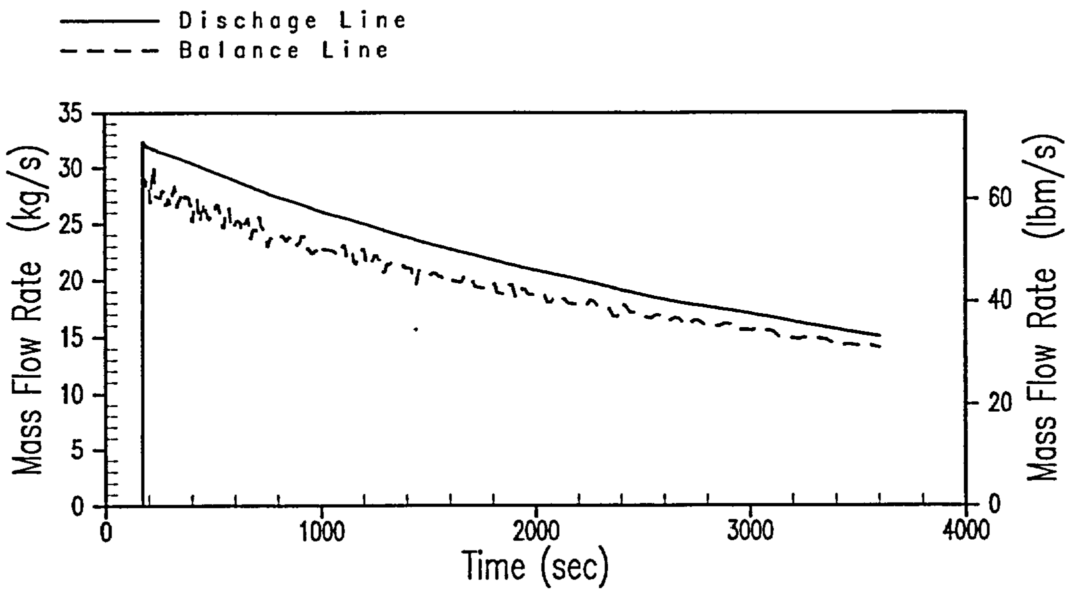


Figure 440.43-5 AP1000 Multiple (5 Tubes) SGTR with PORV Failure CMT Water Mass Flow Rates

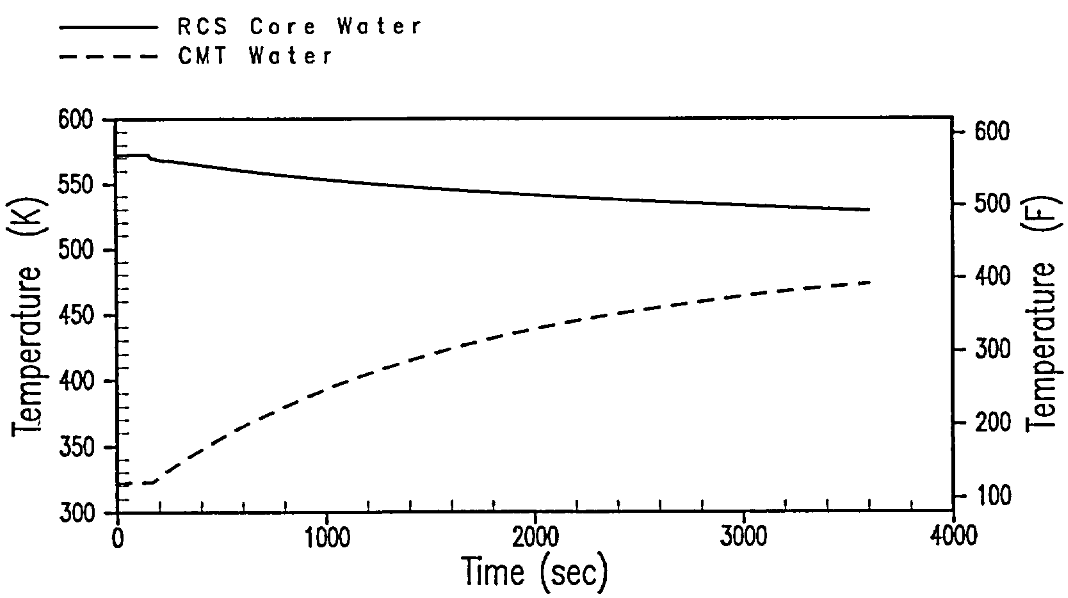


Figure 440.43-6 AP1000 Multiple (5 Tubes) SGTR with PORV Failure RCS and CMT Water Temperatures

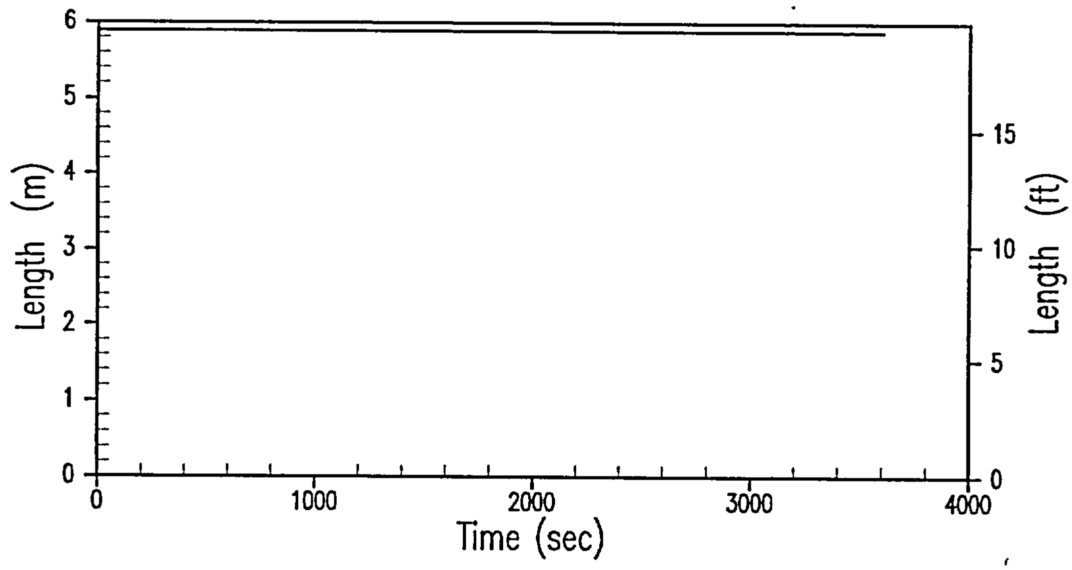


Figure 440.43-7 AP1000 Multiple (5 Tubes) SGTR with PORV Failure CMT Water Level

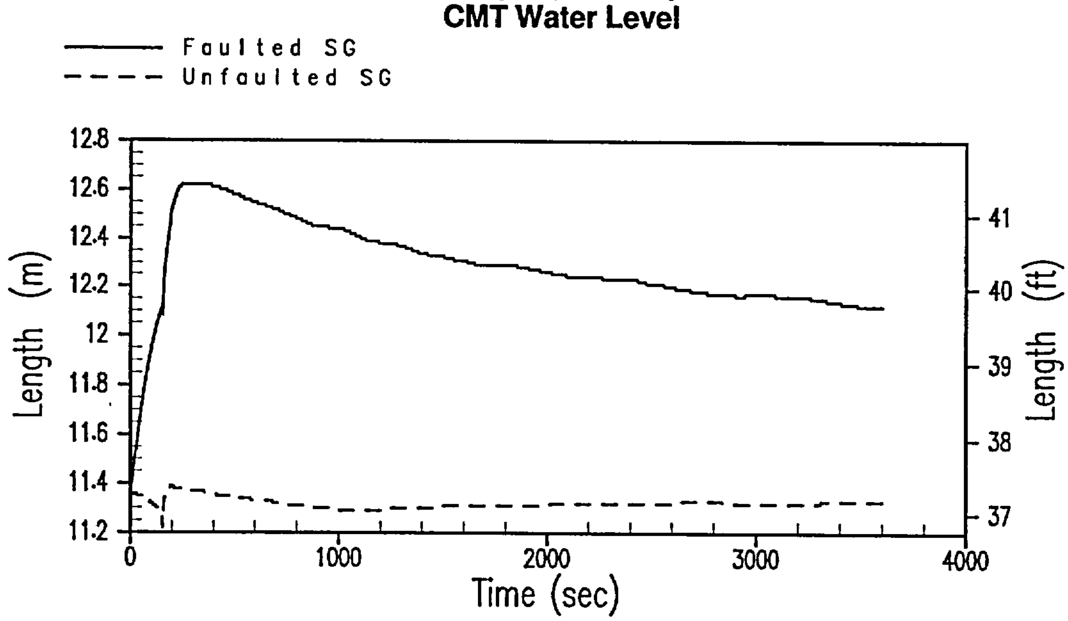


Figure 440.43-8 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Steam Generator Downcomer Water Level

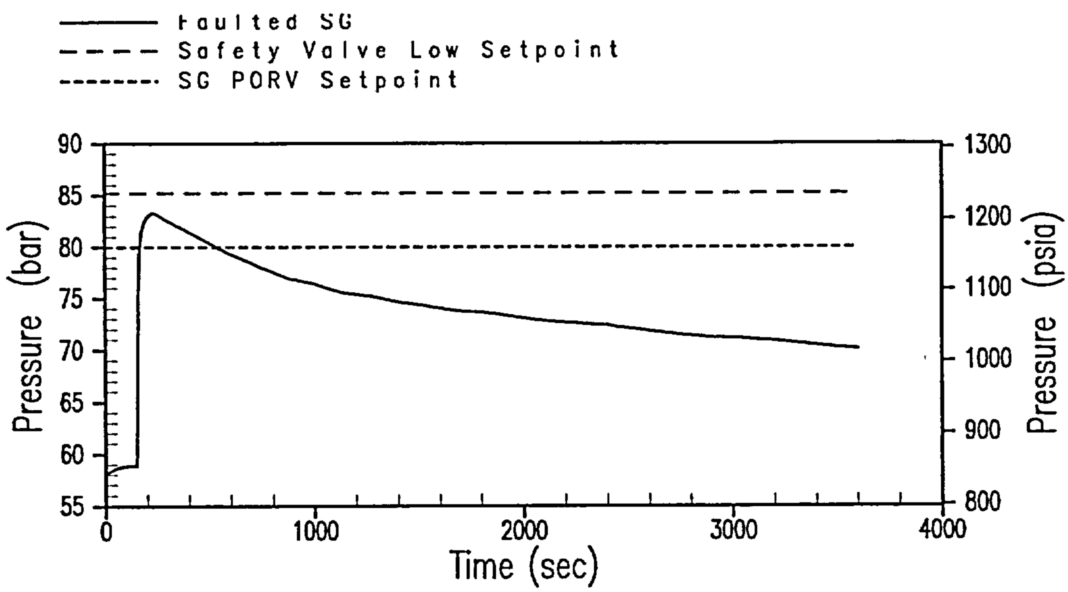


Figure 440.43-9 AP1000 Multiple (5 Tubes) SGTR with PORV Failure Faulted Steam Generator Pressure

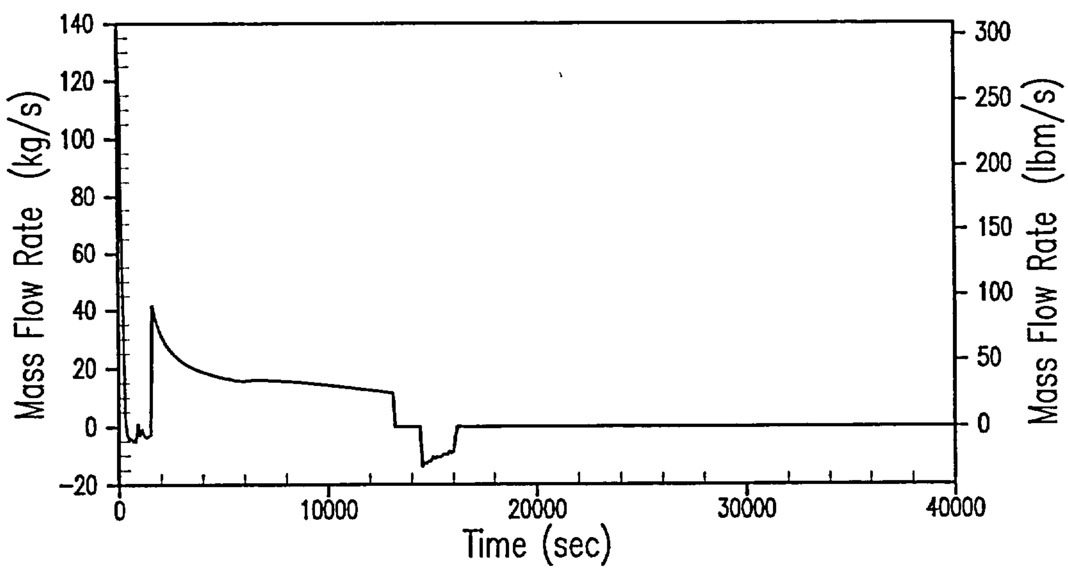


Figure 440.43-10 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Tube Rupture Break Flow

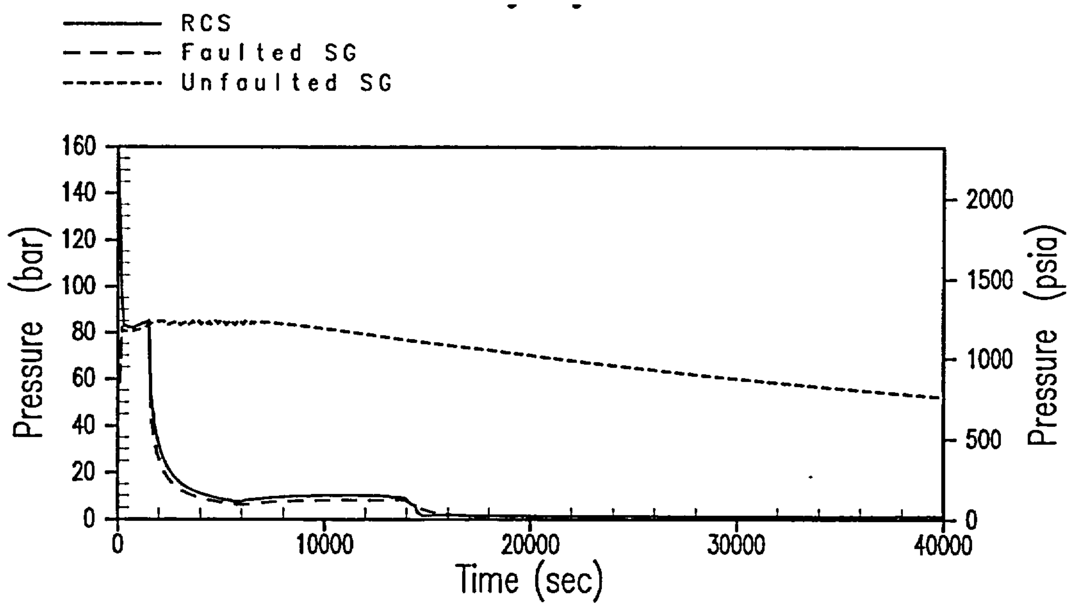


Figure 440.43-11 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS and Secondary System Pressures

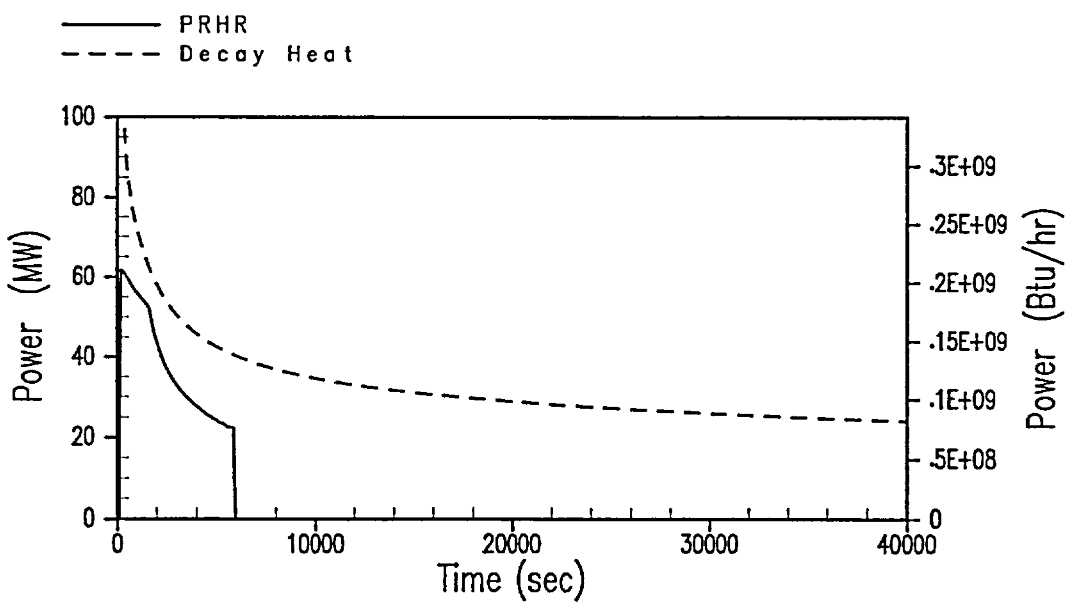


Figure 440.43-12 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve PRHR Heat Removal

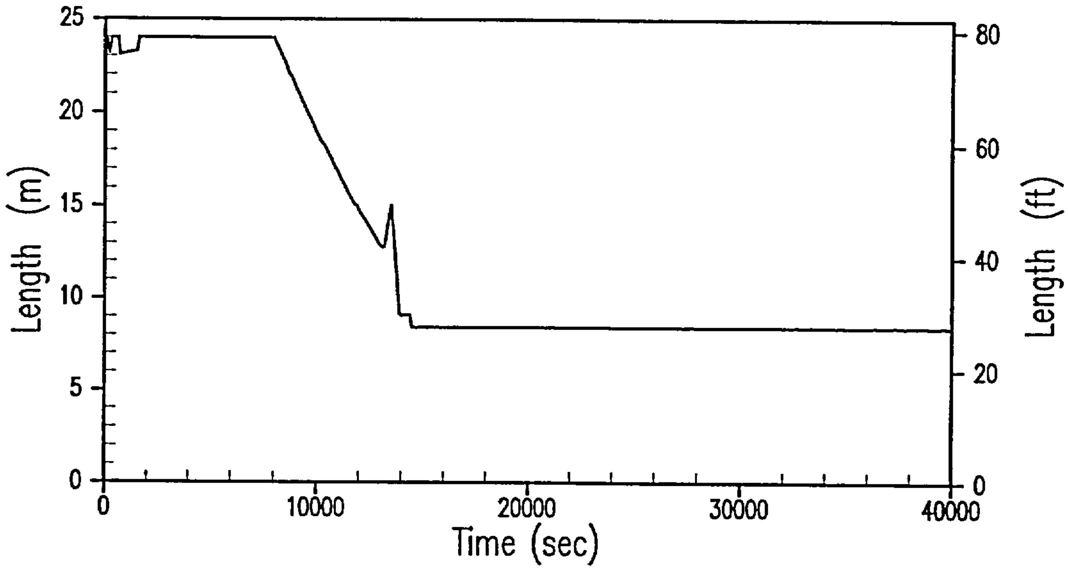


Figure 440.43-13 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS Water Level

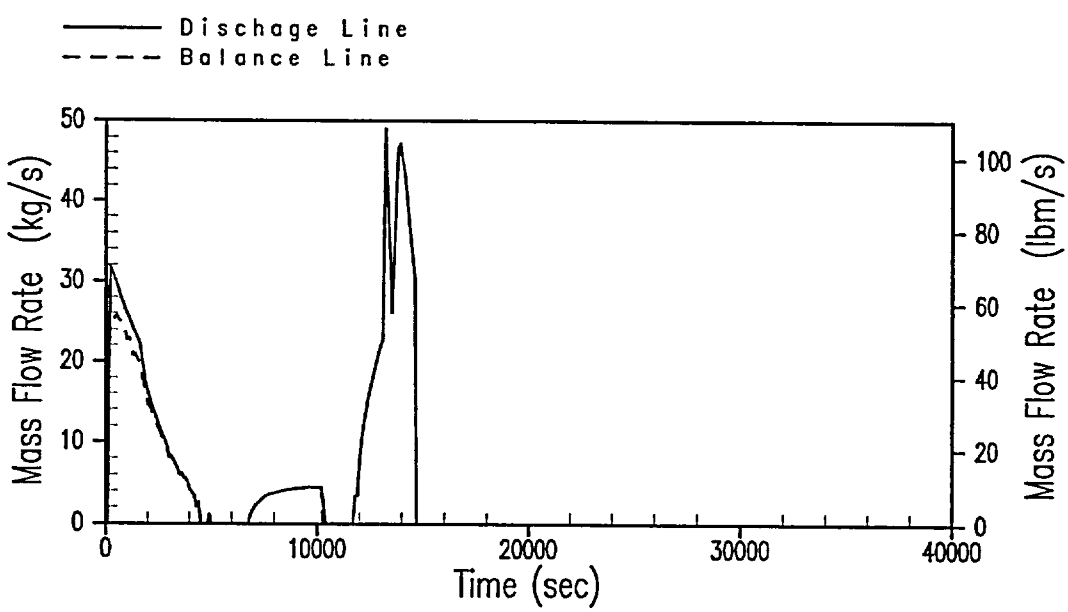


Figure 440.43-14 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve CMT Water Mass Flow Rates

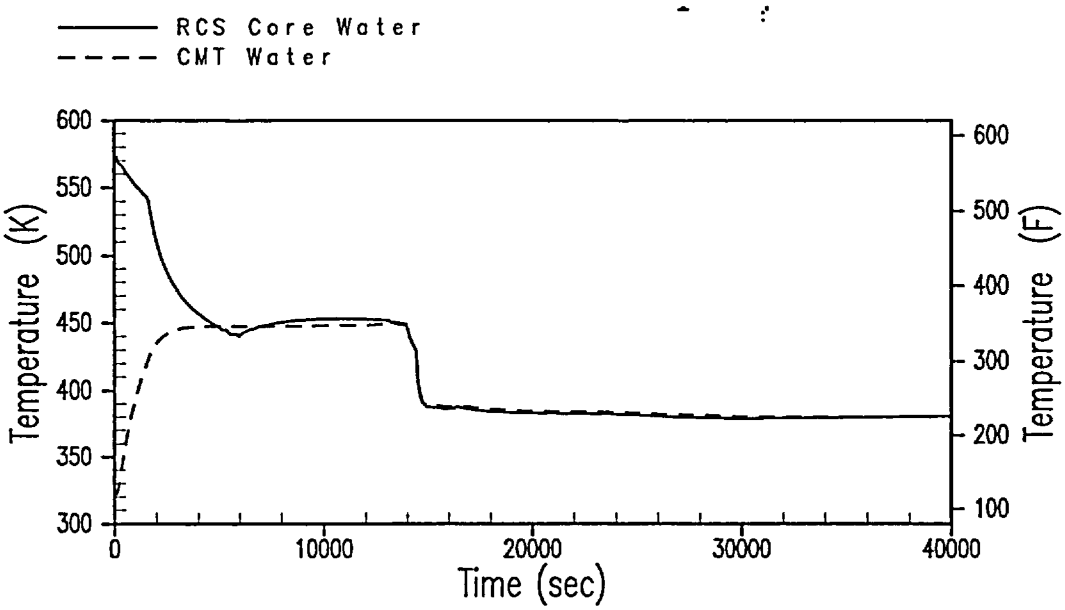


Figure 440.43-15 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve RCS and CMT Water Temperatures

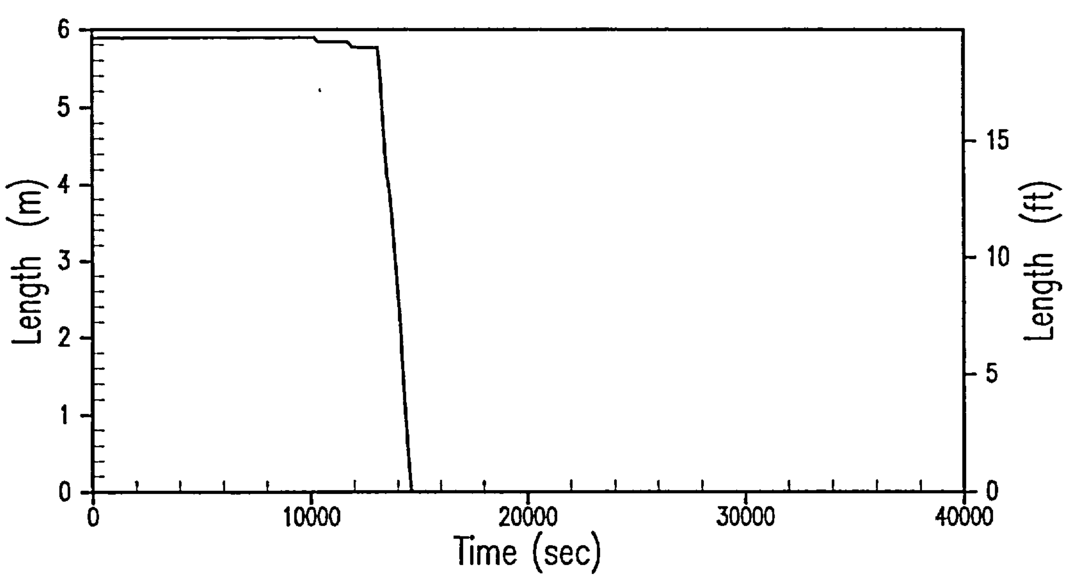


Figure 440.43-16 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve CMT Water Level

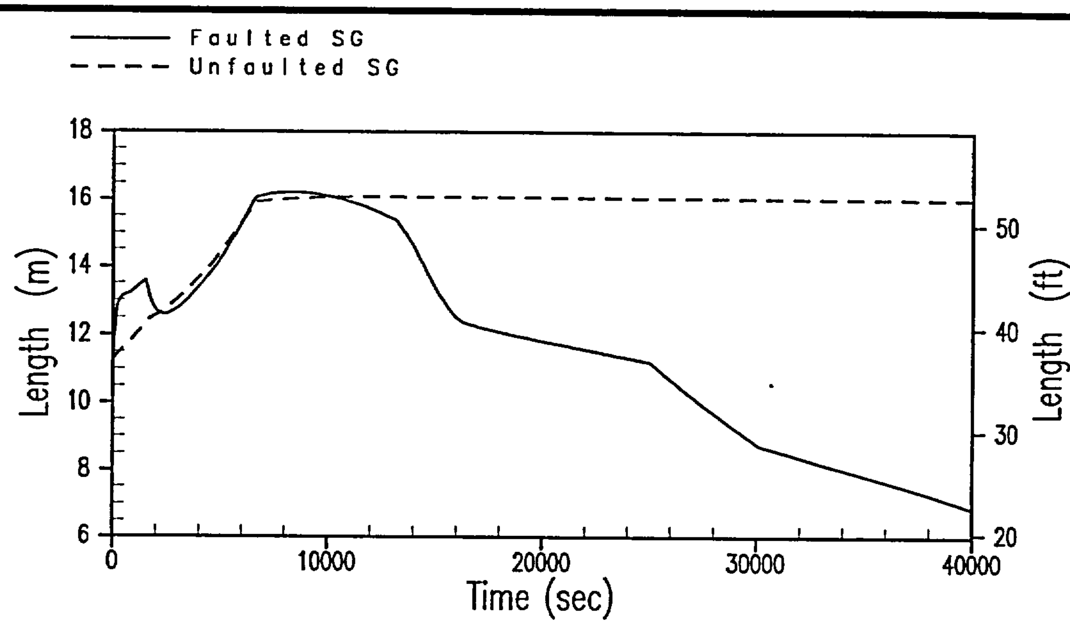


Figure 440.43-17 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Steam Generator Downcomer Water Level

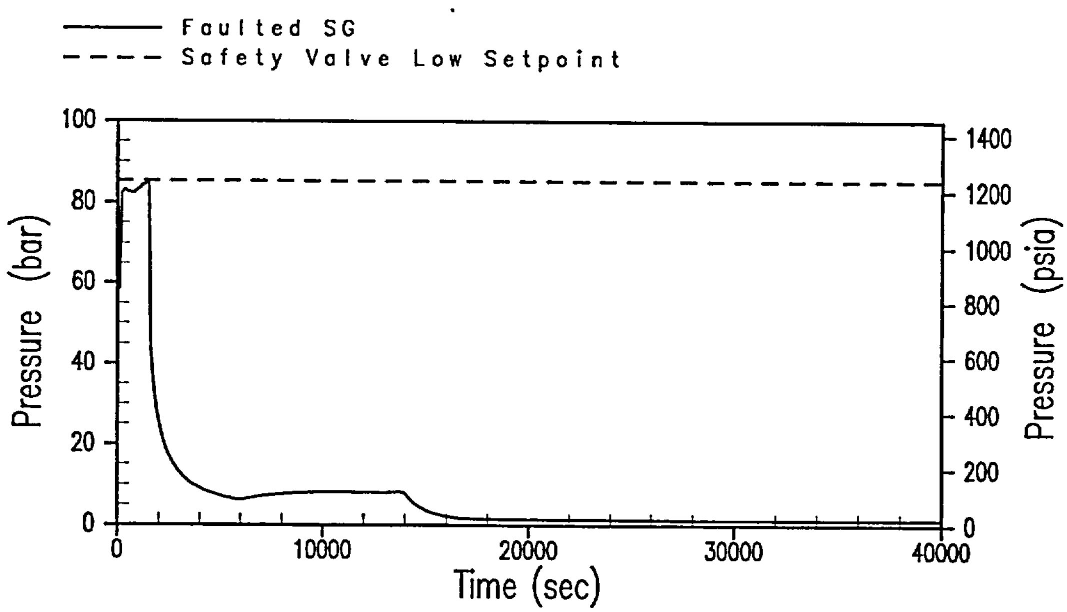


Figure 440.43-18 AP1000 Multiple (5 Tubes) SGTR with Stuck Open SG Safety Valve Faulted Steam Generator Pressure

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.046

Question:

Section 5.4.7 indicates that one of the major functions of the normal residual heat removal system (RNS) is to provide a flow path for long-term post-accident makeup to the containment inventory. Section 5.4.7.1.1 states that this function is a safety-related function and safety design basis, whereas Section 5.4.7.1.2.6 lists this safety-related function under non-safety design bases.

A. Clarify the apparent discrepancy in the categorization of RNS long-term, post-accident containment inventory makeup path function.

B. Describe the RNS design features that satisfy this safety related function.

Westinghouse Response:

The RNS has a safety related function to provide long-term post-accident makeup to the containment to accommodate leakage from the containment. The RNS features used to provide this makeup path are the RNS heat exchanger A channel head drain valve (V046) and the piping and valves to the RCS. The RNS heat exchanger drain valve is a 1" manual valve.

Design Control Document (DCD) Revision:

Delete section 5.4.7.1.2.6 and renumber section 5.4.7.1.2.7.

~~5.4.7.1.2.6 Long-Term, Post-Accident Containment Inventory Makeup Flowpath~~

~~The normal residual heat removal system provides a flow path for long term post-accident makeup to the reactor containment inventory. This capability is provided for long term makeup under design assumptions of containment leakage.~~

5.4.7.1.2.~~7~~6 Spent Fuel Pool Cooling

The normal residual heat removal system has the capability to supplement or take over the cooling of the spent fuel pool when it is not needed for normal shutdown cooling.

Revise section 5.4.7.5, Design Evaluation, to include the long-term post-accident containment makeup function.

5.4.7.5 Design Evaluation

Since the normal residual heat removal system is connected to the reactor coolant system, portions of the system that create the reactor coolant system pressure boundary are designed according to ANSI/ANS 51.1 (Reference 6) with regards to maintaining the reactor coolant system pressure boundary integrity.

Since the normal residual heat removal system penetrates the containment boundary, the containment penetration lines are designed according to the containment isolation criteria identified in subsection 6.2.3.

Safety related makeup water can be provided through the normal residual heat removal system for long-term post-accident containment makeup. This makeup is provided through the manual drain valve in the normal residual heat removal heat exchanger A.

The normal residual heat removal system components and piping are compatible with the radioactive fluids they contain.

The design of the normal residual heat removal system has been compared with the acceptance criteria set forth in subsection 5.4.7, "Residual Heat Removal System," Revision 3, of the NRC's Standard Review Plan. The specific General Design Criteria identified in the Standard Review Plan section are General Design Criteria 2, 4, 5, 19, and 34. Additionally, positions of Regulatory Guides 1.1, 1.29, and 1.68 were also reviewed to determine the degree of compliance between the AP1000 and the acceptance criteria. Branch Technical Position RSB 5-1 was also reviewed as appropriate.

Discussions of the conformance with Regulatory Guides and Branch Technical Positions are found in Section 1.9. Compliance with General Design Criteria is found Section 3.1.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.048

Question:

Section 5.4.12.1 states that the reactor vessel head vent system (RVHVS) is designed to vent a volume of hydrogen at system pressure and temperature equivalent to approximately 40 percent of the RCS volume in one hour.

A. Describe the rationale for appropriateness of this vent capacity design basis.

B. Do the AP1000 RVHVS valves have individual positive valve position indication and alarm in the control room?

Westinghouse Response:

A. DCD Section 5.4.12.1 states the reactor vessel head vent system is sized to vent a volume of hydrogen at system pressure and temperature equivalent to approximately 40% of the reactor coolant system volume in one hour. This is the traditional sizing basis for post Three Mile Island (TMI) reactor vessel head vent systems. However, since the AP1000 utilizes an automatic depressurization system (ADS), collecting of non-condensables in the RV head does not impede core cooling and therefore RV head venting capability is not required for AP1000.

The AP1000 RVHV capacity is also sized to perform the following functions:

- provide an emergency letdown path that can be used to prevent long-term pressurizer overfill following loss of heat sink events.
- normal RCS venting and filling operations during startup.

The limiting requirement is providing an emergency letdown path to prevent long term pressurizer overfill. Capacity of the RVHVS is confirmed by ITAAC 8.e.

From AP1000 ITAACs (DCD Section 2.1.3, Table 2.1.2-4:

8.e) The RCS provides emergency letdown during design basis events.	Inspections of the reactor vessel head vent valves and inlet and outlet piping will be conducted.	A report exists and concludes that the capacity of the reactor vessel head vent is sufficient to pass not less than 8.2 lbm/sec at 1250 psia in the RCS.

B. The AP1000 remotely-operated, safety-related valves, which includes the RVHVS valves (RCS-PL-V150A, B, C, D), have individual positive valve position indication in the Main Control Room. Design of the AP1000 alarm system is part of the HSI design process described in DCD chapter 18.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.056

Question:

Table 15.0-4a lists protection and safety monitoring system setpoints assumed in the accident analysis. These values for the setpoints are included in Table 3.3.1-1 of the Technical Specifications (TS) as trip setpoints for the reactor trip system. However, no total uncertainty allowances are specified for the reactor trips in the TS.

Address the acceptability of the TS for the trip setpoints without inclusion of the instrumentation uncertainties. This question on the TS setpoint uncertainties is also applied to TS Table 3.3.2-1, which specifies the trip setpoints for the engineered safeguards actuation systems without inclusion of the total uncertainty allowances.

Westinghouse Response:

See the response to RAI 440.103 for a discussion of the approach for identifying the allowable values and the trip setpoints for Technical Specifications Tables 3.3.1-1 and 3.3.2-1.

As discussed in the response to RAI 440.103, the Reviewer Note on page 1 of Tables 3.3.1-1 and 3.3.2-1 explain that the AP1000 Technical Specifications only provide the trip setpoint value and do not include the allowable value. Measurement uncertainties for trip and ESFAS instrumentation cannot be determined until the plant specific setpoint calculation is completed by the Combined License applicant, once the actual instrumentation has been selected for the plant. DCD Section 7.1.6 references the setpoint methodology for protection systems that is applicable for the AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.061

Question:

Table 15.0-8 lists the systems or components that are non-safety related but credited in the accident analysis.

A. Discuss, for each of the event category, how these non-safety related equipment are used for mitigation of transients and what are their effects on determination of the limiting case for each event category presented in Chapter 15.

B. Reference the TS sections for these non-safety related equipment to address the compliance of item (c)2(ii)(C) of 10 CFR 50.36 that requires a technical specification (TS) for the systems or components that are used for event mitigation.

Westinghouse Response:

Table 15.0-8 identifies the following non-safety related systems and equipment used for mitigation of design basis analyses.

- trip of the main feedwater pumps
- Closure of MSIV backup valves and main steam branch isolation valves. The MSIV backup valves include the turbine stop & control valves, turbine bypass valves, and moisture separator reheat steam supply control valve.
- pressurizer heater block

Addition of excessive feedwater (DCD Section 15.1.2) causes an increase in core power by decreasing reactor coolant temperature. Overpower reactor trips prevent core thermal limits from being exceeded during excessive feedwater transients. However, the reactor trips do not terminate the excessive feedwater flow. Feedwater flow is terminated by closing the main feedwater isolation valves, closing the main feedwater control valves and tripping of the main feedwater pumps. An excessive feedwater event can be postulated due to a fault in the main feedwater control valve. As a consequence of the initiating fault, it could also be postulated that the main feedwater control valve would fail to close on demand. If as an independent single failure, the main feedwater isolation valve on the loop with the faulted feedwater control valve also fails to close, the non-safety related feedwater pump trip is relied upon to terminate main feedwater flow and prevent overfilling of the steam generator.

The MSIV backup valves and the main steam branch isolation valves are credited to close during steam generator depressurization transients due to the inadvertent opening of a secondary relief valve or due to steam line ruptures (DCD Sections 15.1.4 and 15.1.5). Crediting these valves to close on demand prevents the uncontrolled blow down of two steam generators in the event of an independent single failure of the MSIV in the intact steam generator to close (see also the response to RAI No. 440.069 for further discussion).

One of the acceptance criteria for faults of moderate frequency (Condition II) is that these events do not propagate to a more serious fault (i.e. Condition III or IV). To address this criterion, the acceptance criterion of no filling of the pressurizer is assumed for moderate frequency events. Adoption of this criterion prevents water relief through the pressurizer safety valves. Loss of normal feedwater (DCD Section 15.2.7), inadvertent operation of the core makeup tanks (DCD Section 15.5.1) or inadvertent operation of the chemical and volume control system (DCD Section 15.5.2) can result in pressurizer water level increases that may challenge filling of the pressurizer. Operation of the pressurizer heaters during these events causes expansion of the fluid within the pressurizer. The pressurizer heater block is assumed to prevent undesired pressurizer heater operation from exacerbating the expansion of RCS fluid during these events.

Mitigation of steam generator tube rupture transients (DCD Section 16.6.3) is accomplished by promoting the reaching of an equilibrium pressure in the reactor coolant and the faulted steam generator. Operation of the pressurizer heaters would tend to keep the reactor coolant pressure higher. This would induce additional primary to secondary leakage. The automatic pressurizer heat block terminates the pressurizer heaters without the need for additional operator action. Similarly, allowing the faulted steam generator to depressurize through the turbine would increase the primary to secondary fluid leakage. Closure of MSIV backup valves and main steam branch isolation valves prevents the uncontrolled depressurization of the faulted steam generator.

During small break LOCA events, the reactor coolant system is depressurized using ADS. The first three stages of the ADS are connected to the pressurizer. During a depressurization with the ADS the pressurizer level increases and the pressurizer is filled. The addition of energy from the pressurizer heaters would tend to retard the rate of depressurization of the RCS. The pressurizer heaters would also add additional energy to the fluid being released through Stage 1, 2 & 3 of the ADS. The pressurizer heater block prevents interactions between the heaters and ADS and prevents the pressurizer heaters from retarding the rate of RCS depressurization.

Limiting conditions for operation have been established in the AP1000 technical specifications for the non-safety related equipment assumed for mitigation of the design basis events and the use of these non-safety related systems and components comply with Item (c)2(ii)(C) of 10 CFR 50.36. Item 7 of Table 3.3.2-1 of the technical specifications includes a surveillance requirement for the feedwater pump trip. Item 27 of Table 3.3.2-1 of the technical specifications includes a surveillance requirement for the pressurizer heater trip. Limiting conditions for operation for the backup main steam isolation valves and the main steam branch isolation valves are provided in section 3.7.2 of the technical specifications.

In Section 15.2 of NUREG-1512 (Reference 1) during the AP600 review, the staff concluded that crediting these non-safety related backup protection systems and components was acceptable because of the following.

(1) The trip mechanisms of the feedwater pump trip breakers and pressurizer heater trip breakers are simple, and the likelihood of the breaker function failure is low.
(2) The operating data show that the turbine stop and control valves are reliable, and taking credit of the turbine valves in the design-basis analyses for backup protection is consistent with the staff position stated in NUREG-0138 (Reference 2)
(3) The AP600 included surveillance requirements and limiting conditions for operation in the technical specifications to ensure the reliability of the following systems or components:
    (i) feedwater pump trip breakers and redundant pressurizer heater trip breakers
    (ii) the main steam branch isolation valves
    (iii) the MSIV backup valves

References

1) NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design Docket No. 52-003," September 1998

2) NUREG-0138, "Staff Decision of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff."

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.062

Question:

Westinghouse has issued three Nuclear Service Advisory Letter (NSAL), NSAL-02-3 and revision 1, 02-4 and 02-5, which document the problems with the Westinghouse-designed steam generator (SG) water level setpoint uncertainties. NSAL-02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break.) NSAL-02-4, issued on February 19, 2002, deals with the uncertainties created because the void contents of the two-phase mixture above the mid-deck plate were not reflected in the calculation and affect the high-high level trip setpoint. NSAL-02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate pressure differential pressure which have resulted in significant increases in the control system uncertainties.

A. Discuss how the AP1000 design accounts for all of these uncertainties documented in these advisory letters in determining the SG water level setpoints.

B. Discuss the effects of the water level uncertainties on the analyses of the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the analyses used to support the design certification are still limiting.

Westinghouse Response:

Measurement uncertainties for trip and ESFAS instrumentation cannot be determined until actual instrumentation has been selected for the plant. The plant specific setpoint calculations will be completed and reviewed as part of the Combined Operating License. Combined License applicants referencing the AP1000 certified design will provide a calculation of setpoints for protective functions consistent with the methodology presented in Reference 1. Reference 1 is an AP600 document that describes a methodology that is applicable to AP1000. Applicable uncertainties documented in the referenced Nuclear Services Advisory Letters will be incorporated as needed when the setpoint study is performed.

Plant nominal setpoints are calculated by adding the channel allowance from the setpoint study to the setpoints used in the safety analysis. In performing the safety analyses, conservative safety analysis setpoints are chosen for the protection system setpoints assumed. The setpoints used for the safety analyses are chosen such that when channel allowance are added to the safety analysis setpoint, viable nominal setpoints will result.

References

1. WCAP-14605 (P), WCAP-14606 (NP), "Westinghouse Setpoint Methodology for Protection Systems, AP600," April 1996.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.069

Question:

Item d of the acceptance criteria in SRP 15.1.5 states that "the worst single active component failure should be assumed to occur. The assumed single failure may cause more than one steam generator to blow down, or may be in any of the systems required to control the transient."

Discuss the determination of the worst single active failure and address its compliance with the SRP guidance related to the SLB with steam blowdown from both steam generators.

Westinghouse Response:

A steam line break resulting in steam blowdown from both steam generators is precluded by the AP1000 design. In addition to closing of the main steam isolation valve (MSIV) in each steam line, the non-safety related turbine stop valves, turbine control valves and main steam branch isolation valves are closed to preclude the blowdown of a second steam generator in the event that the safety-related main steam isolation valve in the intact steam generator fails to close. Backup isolation crediting closure of the turbine stop valves, the turbine control valves and the main steam branch isolation valves is consistent with the NRC positions taken in Section 10.3 of the SRP (Reference 1) and in NUREG-0138 (Reference 2).

SRP Section 15.1.5 also states in Item 2 of the review procedures subsection,.

> *"To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. This phase of the review is done using the systems review procedures described in the SRP sections for Chapters 5, 6, 7, 8, and 10 of the SAR. The reviewer also considers single failures that may cause more than one steam generator to blow down, thus increasing the reactivity addition to the core. "*

The SRP section for Chapter 10 (specifically Section 10.3) references NUREG-0138. Technical Issue 1 of NUREG-0138 discusses treatment of non-safety related equipment in steam line break transients and identifies a steam line break scenario where the break is un-isolatable and is assumed to be between the steam generator and the MSIV. If as a single failure, the MSIV in the intact steam generator loop fails to close, the intact steam generator could continue to blow down through the turbine and main steam system branch lines if no further action is taken. SRP Section 10.3 and NUREG-0138 conclude that reliance on non-safety related valves as backup isolation is permitted to prevent this scenario.

The AP1000 primary main steam isolation function is performed by the closure of the safety-related main steam isolation valves. There is one MSIV per main steam line. As a backup, non-safety-related valves downstream of the MSIV are automatically closed on various safety-related signals from the protection system.

On a turbine trip signal (generated by any reactor trip signal, Safeguards signal, or High-2 steam generator level signal), the following non-safety-related main steam valves are closed to provide a backup isolation function:

- Turbine stop and control valves (the turbine stop valves and control valves are in series and provide backup protection to one another.)

- Main steam branch isolation valves

For steam line breaks occurring upstream of the MSIVs, the backup steam isolation valves prevent a second steam generator from blowing down through the turbine or a branch line in the event that the MSIV on the unfaulted steam generator fails to close on demand. No single active failure will result in blowdown of both steam generators for the AP1000.

Mitigation of AP1000 steam line breaks is also accomplished by tripping the reactor, borating of the reactor coolant system with automatic actuation of the core makeup tanks (CMTs) and automatic isolation of feedwater. The reactor trip functions used for steam line break are single failure proof.

Feedwater isolation is also single failure proof. Feedwater isolation is accomplished by closure of safety related main feedwater isolation valves. The protection system feedwater isolation signals also activate backup feedwater isolation by closure of non-safety related feedwater control valves and tripping of the main feedwater pumps. The use of the non-safety related feedwater control valves and the main feedwater pump trip is consistent with the positions established in SRP Section 10.3 and NUREG-0138.

The limiting single failure for AP1000 steam line breaks is the failure of a redundant parallel CMT discharge valve to open. This failure is assumed in Section 15.1.5 of the AP1000 DCD. This failure will result lower CMT flow rates due to higher CMT injection line pressure loss coefficients. The failure of a CMT discharge valve to open does not cause the CMT to be inoperable.

References

1) NUREG-0800, " Standard Review Plan for the Review of safety analysis Reports for Nuclear Power Plants"

2) NUREG-0138, "Staff Discussion of Fifteen Technical issues listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff", November 1976

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.070

Question:

It states on page 15.1-16 of Chapter 15 that for the SLB analysis, power peaking factor corresponding to one stuck RCCA is determined at the end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod.

Discuss the methods used for the power peaking factor determination and address the acceptance of the methods and computer codes used. Provide values of the calculated total power peaking factors and justify that they are conservative for calculations of the minimum DNBRs during an SLB event.

Westinghouse Response:

The consequences during the main steamline rupture are evaluated at the end of cycle operation where the largest reactivity insertion is expected due to the moderator cooldown caused by excessive secondary side steam flow. The cooldown can be maximized when no heat source is available in the core. Thus, the reactor is assumed to be in hot zero power with no decay heat prior to initiation of the accident.

A detailed description of the accident is presented in DCD Section 15.1.5. Shortly after the accident initiation, the reactor is tripped, i.e., all control banks are inserted into the reactor core. However, for conservatism in the analysis, a RCC of the highest worth is assumed to be stuck out of the core. For the AP1000 reference core, the highest worth RCC are found to be at location J-3 (one RCC of the second shutdown bank). This is defined the worst stuck RCC. The basic core configuration for the steamline rupture analysis is at EOC with all RCCs inserted except the worst stuck RCC. This RCC configuration is referred to as "N-1 rodded condition". When the reactor is shutdown at hot zero power with the N-1 RCCs inserted, the reactor is usually subcritical more than the design shutdown margin (SDM). For the AP1000 core design, the SDM is defined as 1.6%. Again for conservatism, an assumption is made that the reactor is just to be subcritical by the SDM when the N-1 RCCs are inserted.

The LOFTRAN code is used to analyze the transient. The LOFTRAN reactor model was generated to reproduce reactor conditions anticipated during the accident, analyzed by the licensed design codes, such as ANC and VIPRE-01 (Reference 57 of DCD Section 4.3, Reference 22 and Reference 83 of DCD Section 4.4). The LOFTRAN reactor model is described in detail in Reference 440.070-1.

The accident analysis was performed using the LOFTRAN code as described in DCD Section 15.1.5. The results of the analysis are shown in DCD Figures 15.1.5-1 through 15.1.5-13.

In order to verify the accuracy and conservatism of the LOFTRAN analysis, the consistency of the reactivity behavior throughout the transient between the LOFTRAN and the three-dimensional neutronic/thermal-hydraulic analysis by the design codes was demonstrated. These were performed at selected time steps during the transients. These are called "Statepoints". Three statepoints were selected and reactor conditions obtained by LOFTRAN are summarized in Table 440.070-1.

For three selected statepoints, the reactor power is searched as follows:

I. The reactor is at EOL, with hot full power equilibrium xenon.

II. The shut down reactivity is defined at hot zero power with N-1 RCCs inserted.

III. The shut down reactivity is increased by the design shutdown margin. This is the eigenvalue at which the 3D-core model is considered to be at critical condition. This is called the "target" eigenvalue.

At each state point, the inlet temperature, pressure, boron concentration and coolant flow rate are input, then the reactor power is searched for the target eigenvalue. Here again for conservatism, the affected loop inlet temperature is used for the entire assemblies. Due to low coolant flowrate, the water density distribution is calculated by the VIPRE-01 code taking the cross flow among the fuel channels into account. The cross flow modeling of the Westinghouse design code has been qualified by the flow temperature and velocity measurements within an electrically heated rod bundle containing steep radial power gradient performed at the (Battelle) Pacific Northwest Laboratory (PNL). The qualification results are documented in Reference 440.070-2.

The ANC/VIPRE-01 searched power is also shown in Table 440.070-1. The acceptance criteria is that the searched power should be close and slightly above the LOFTRAN power, typically within a couple of percent of power. The reasoning is that overestimating the searched power comes from overestimating the cooldown effect, which is in a conservative direction for the statepoint power prediction by the neutronic code. Table 440.070-1 shows that comparison of the LOFTRAN and searched power behaves very favorably. From this study, the LOFTRAN transient is confirmed to be consistent with the reactor core design code accuracy.

The nuclear fuel integrity is evaluated at the most limiting statepoint, i.e., statepoint No 2 in Table 440.070-1. The radial and axial power distribution is shown in Figure 440.070-1 and Figure 440.070-2. The DNBR calculation was performed for the power distribution shown in Figure 440.070-2. It was confirmed that the design criterion was met with a comfortable margin in spite of several conservative assumptions used throughout the calculational process as mentioned above.

References:

440.070-1 "AP1000 Analysis Methodology Summary for Events Using the LOFTRAN Code Family", APP-GW-GSR-010, Appendix A Transmitted via Letter DCP/NRC 1488, Oct. 31, 2001, to A.C. Rae(NRC) from M. M. Corletti(Westinghouse) as "Transmittal of Westinghouse Report"

440.070-2 Morita,T., et al., "Subchannel Thermal-Hydraulic Analysis at AP600 Low-Flow Steam-line Break Conditions", Nuclear Technology, Vol, 112, P.401, December, 1995, American Nuclear Society

Design Control Document (DCD) Revision:

None

PRA Revision:

None

TABLE 440.070-1

AP1000 Steamline Rupture Transient StatePoint

t (sec)	T_{In} Cold (°F)	Pressure (psia)	Inlet Flow (%)	C_B (ppm)	Reactor Power (%)	
					LOFTRAN	ANC/VIPRE
100	416.6	943.6	8.0	73.0	1.1	2.0
240 (Max Power)	327.6	684.0	6.0	138.0	3.3	4.7
400	291.6	478.6	5.0	265.0	1.9	2.3

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Response to Request For Additional Information

FIGURE 440.070-1

AP1000 Assembly wise FΔH Distribution
at
Maximum Power StatePoint during Maximum Steamline Rupture Transient

180

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1								3.858	3.629	3.475							
2						1.884	3.167	6.101	5.353	5.278	1.830	1.090					
3					1.415	2.047	4.763	5.020	* 6.202	2.982	2.478	1.165	0.911				
4				1.067	1.951	2.552	2.446	4.546	2.679	2.847	1.224	1.623	1.303	0.754			
5			0.649	1.060	1.962	1.303	1.629	1.497	2.379	1.208	1.090	0.890	1.391	0.755	0.523		
6			0.743	1.220	0.791	1.028	1.233	2.005	1.425	1.845	1.039	0.842	0.603	1.007	0.646		
7		1.202	1.824	0.989	1.154	0.740	1.586	0.970	1.677	0.926	1.416	0.660	1.000	0.892	1.651	1.095	
8	90	1.120	1.655	1.777	0.662	1.252	0.964	1.346	0.574	1.273	0.882	1.147	0.598	1.614	1.505	1.034	270
9		1.183	1.777	0.963	1.014	0.653	1.231	0.695	1.143	0.657	1.165	0.608	0.931	0.882	1.625	1.079	
10			0.656	0.958	0.492	0.656	0.734	1.120	0.761	1.101	0.696	0.621	0.457	0.888	0.604		
11			0.462	0.529	0.929	0.547	0.614	0.557	1.007	0.553	0.602	0.527	0.891	0.503	0.432		
12				0.513	0.781	0.908	0.447	0.857	0.538	0.853	0.438	0.881	0.753	0.491			
13					0.498	0.503	0.836	0.807	1.466	0.805	0.830	0.490	0.484				
14						0.415	0.560	1.491	1.368	1.493	0.559	0.410					
15								0.991	0.940	0.995							

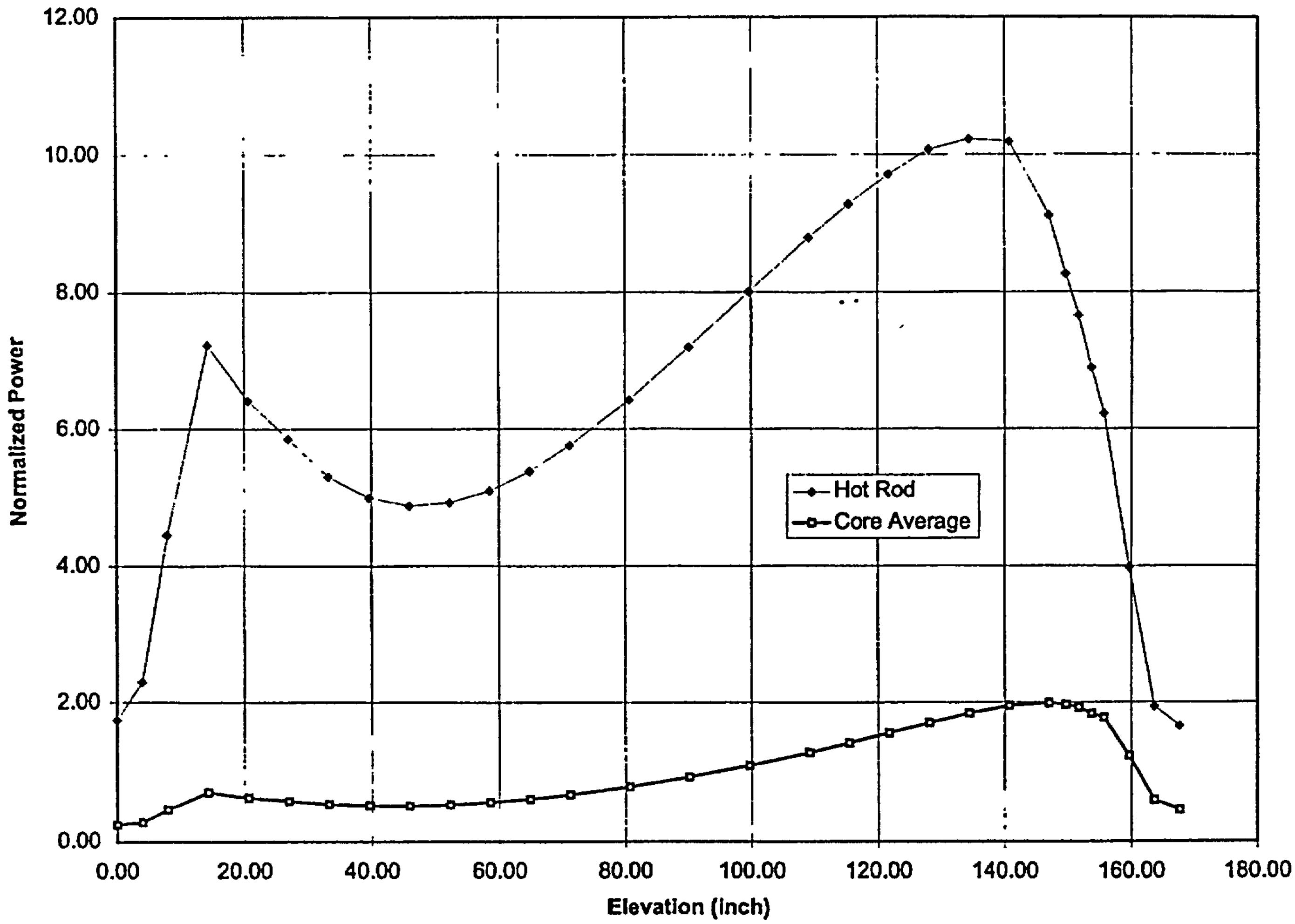
0

: Stuck RCC with highest worth

* : Max FΔH

FIGURE 440.070-2

AP1000 Axial Power Distribution



RAI Number: 440.080

Question:

Section 15.3.3 presents the results of analysis for the locked rotor event. The analysis assumes that DNB starts to occur at the beginning of the accident. In accordance with the SRP 4.4 guidance, all fuel rods experiencing DNB should be assumed to fail.

Calculate the number of failed fuel rods in accordance with the SRP 4.4 guidance and verify that the calculated failed rod number is within the assumed value used for the radiological calculations.

Westinghouse Response:

In performing analyses for the locked rotor event, several distinct analyses are performed emphasizing different challenges to acceptance criteria. One analysis is performed to calculate peak RCS pressure. Another analysis is performed to calculate the minimum DNB ratio, which can be obtained for the number of rods with a DNB ratio below the acceptance limit. A third analysis is performed to maximize the clad temperature and the amount of zirconium-steam reaction. In these three analyses, different assumptions and initial conditions are selected to maximize or minimize the quantity of interest with respect to the acceptance criteria.

First, LOFTRAN is used to calculate the overall system response and calculate the nuclear power. Then, using the transient nuclear power from LOFTRAN, the FACTRAN code is used to calculate fuel rod thermal power transients. Three separate FACTRAN cases are performed: fuel rod average channel, fuel rod hot channel and fuel rod hot spot. The fuel rod heat fluxes from the fuel rod average channel and hot channel cases are used as input to the VIPRE-01 code for the purpose of calculating DNBR.

The third FACTRAN case, at the fuel rod hot spot, is performed for the purpose of calculating the maximum clad temperature. In this third FACTRAN analysis, DNB is assumed to occur at the beginning of the event. By assuming DNB at the beginning of the event, the code is forced to calculate the heat flux using a film boiling heat transfer coefficient. This maximizes the clad temperature and maximizes the amount of zirconium-steam reaction which occurs.

The assumption that DNB occurs at the beginning of the transient is not used in the calculations for the minimum DNBR for the number of fuel rods that exceed the DNBR limit. For the locked rotor event, the WRB-2M correlation is applicable to the local conditions at the time of minimum DNBR. The minimum DNBR is above the DNBR Design Basis and no rods fail.

Although no fuel rods were assumed to fail, the calculations for dose releases were performed assuming 16% of the fuel rods were damaged. The response to RAI Number 470.002 discusses the basis for assuming 16% fuel rod damage for the locked rotor dose analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.082

Question:

Section 15.4.3.3 indicates that for cases of dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA, the calculated DNBRs remain greater than the safety DNBR limits. However, no calculated DNBRs are presented.

Provide figures showing calculated DNBRs during the transients and demonstrate that no fuel rod is predicted to fail for the events of RCCA drop and RCCA misalignment.

Westinghouse Response:

The DROPROD code is used to confirm that DNBR limits are met in dropped RCCA and dropped RCCA bank scenarios. For a large number of limiting transient statepoints, this code determines how high the pre-drop F-delta-H would need to be in order for the dropped rod event to cause rods to be in DNB (higher pre-drop F-delta-H values give higher post-drop F-delta-H, and hence lower DNBR, values). Demonstrating that the limiting pre-drop value cannot be achieved in normal operation confirms that DNB will not occur during the dropped rod event. This has been demonstrated for the AP1000. Post-drop F-delta-H and DNBR values are not calculated explicitly for this event.

For the Static Rod Misalignment event, the calculated F-delta-H (including uncertainty) is 1.91 compared to the limit of 2.07. Using the sensitivity of DNBR to F-delta-H at Static Rod Misalignment conditions of 1.0% F-delta-H / 2.5% DNBR, the 8% F-delta-H margin to the limit is equivalent to 20% DNB margin.

Design Control Document (DCD) Revision:

None

PRA Revision:

None